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Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement, Volume 1, Appendix D, Part B

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Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement

Volume 1
Appendix D
Part B

Naval Spent Nuclear Fuel Management

April 1995

U.S. Department of Energy
Office of Environmental Management
Idaho Operations Office

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ATTACHMENT A - TRANSPORTATION OF NAVAL SPENT NUCLEAR FUEL

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont)

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

ATTACHMENT A

TRANSPORTATION OF NAVAL SPENT NUCLEAR FUEL

A.1 PURPOSE AND SCOPE

This attachment provides an evaluation of the radiological and non-radiological risks associated with the transportation of naval spent nuclear fuel and test specimens that originate from Navy and commercial shipyards, prototypes, and related Department of Energy laboratories. This evaluation covers all past shipments through May 1995 and shipments planned in the 40-year period from June 1995 through the end of 2035. This attachment evaluates the radiological risks associated with the five alternatives described in Section 3.

A.2 BACKGROUND

The transportation of naval spent nuclear fuel and test specimens covered in this attachment falls into the following four categories:

- Shipments of Naval Spent Nuclear Fuel from Shipyards and Prototypes
- Transfers of Naval Spent Nuclear Fuel to Storage Following Examination
- Transfers of Naval Test Specimen Assemblies Between the Examination Facility and the Test Reactor Area
- Shipments of Naval Test Specimens to Examination and Testing Facilities.

Each category is described in more detail below.

A.2.1 Shipments of Naval Spent Nuclear Fuel from Shipyards and Prototypes

Since 1956, spent nuclear fuel has been removed from Navy nuclear-powered ships and prototypes as a routine part of their operational cycle. The spent nuclear fuel has been transported to the Expended Core Facility (ECF) for examination and evaluation. ECF is part of the Naval Reactors Facility (NRF) within the Idaho National Engineering Laboratory (INEL). The examinations of the spent nuclear fuel and irradiated test specimens have provided and will continue to provide engineering data for materials and designs used in technology development for naval nuclear reactors.

In the past, shipments have originated from two prototype sites, nine shipyard locations, and the Shippingport Atomic Power Station (SAPS), located in Shippingport, Pennsylvania. The two prototype locations are the Kenneth A. Kesselring Site (KSO), located in West Milton, New York and the Windsor Site Operation (WSO), located in Windsor, Connecticut. The nine shipyard locations are Newport News Shipbuilding (NNS), located in Newport News, Virginia; the Norfolk Naval Shipyard (NOR), located in Portsmouth, Virginia; the Pearl Harbor Naval Shipyard (PHNS), located in Pearl Harbor, Hawaii; the Portsmouth Naval Shipyard (PNS), located in Kittery, Maine; the Puget Sound Naval Shipyard (PSNS), located in Bremerton, Washington; the Charleston Naval Shipyard (CNS), located in Charleston, South Carolina; the Mare Island Naval Shipyard (MINS), located in Vallejo, California; the Electric Boat Division of General Dynamics (EB), located in Groton, Connecticut, and Ingalls Shipbuilding (INGL), located in Pascagoula, Mississippi. Figure A-1 provides a map of the United States showing the transportation origins for naval spent nuclear fuel. No future shipments from the Electric Boat Division, Ingalls Shipbuilding, and Shippingport Atomic Power Station facilities are planned. The Mare Island Naval Shipyard, Charleston Naval Shipyard, and Windsor Site Operations facilities are being phased out.

The naval spent nuclear fuel has been shipped in M-130, M-140, M-160, and S2W/S2Wa shipping containers. Only the M-130, M-140, and M-160 shipping containers will be used in the future. A detailed description of the shipping containers to be used for naval spent nuclear fuel shipments from shipyards and prototype sites is provided in Section A.4.1.

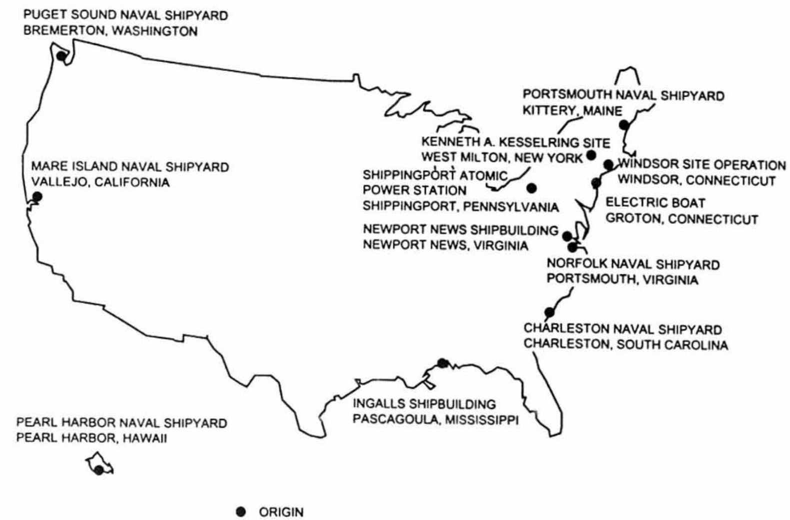

Figure A-1. Transportation origins for naval spent nuclear fuel.

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The naval spent nuclear fuel is primarily shipped by rail. However, for the two prototype sites, rail spurs to the sites are not available. Therefore, the shipping containers are transported by heavy-lift transporter to a nearby commercial rail line where the containers are then transported by rail. For the Pearl Harbor Naval Shipyard, the containers are transported by ship to the Puget Sound Naval Shipyard where the containers are then transported to ECF by rail. Since 1956, 599 containers of naval spent nuclear fuel have been shipped to ECF. An additional 16 containers of spent nuclear fuel were shipped (12 from Shippingport Atomic Power Station to Hanford and 4 from ECF to Hanford); however, these shipments are covered by the DOE historic shipment calculations in Appendix I, Volume 1 of this Environmental Impact Statement. Table A-1 provides a list of these shipments made by year and originating facility.

A.2.2 Transfers of Naval Spent Nuclear Fuel to Storage Following Examination

In the past, following examinations at ECF, the spent nuclear fuel has been prepared and transferred to the Idaho Chemical Processing Plant (ICPP), also located on the INEL. A detailed description of the operations performed in the Expended Core Facility is provided in Attachment B. Naval spent nuclear fuel is currently being held at ICPP until permanent disposition becomes possible.

Since 1956, approximately 5400 transfers of naval spent nuclear fuel have been made from ECF to ICPP in shipping casks transported by truck dedicated to performing only such shipments (exclusive-use). For alternatives involving continued transfers to storage, the transfers would be made in the NFS-100, Peach Bottom, and Large Cell casks, in exclusive-use trucks. A detailed description of the shipping casks used for naval spent nuclear fuel transfers to storage is provided in Section A.4.2.

A.2.3 Transfers of Naval Test Specimen Assemblies Between the Examination Facility and the Test Reactor Area

In addition to naval spent nuclear fuel from ships and prototypes, irradiated test specimen assemblies (fuel and non-fuel) have also been transported to ECF for examination. Test specimens, which are constructed of plant materials, reactor structural materials, and fuels used in naval reactor

Table A-1. Number of past naval spent nuclear fuel containers shipped to ECF by origin.

	Origin												
Year	EB	SAPS	KSO	MINS	PHNS	PSNS	NNS	PNS	CNS	WSO	NOR	INGL	TOTAL
1957	1												1
1958			1										1
1959	1							1					2
1960													0
1961	1	2	2										5
1962	5			1	1								7
1963		3		1	1								5
1964	2	1	2										5
1965	2	1		2			33	1	2		1		42
1966	4	2		1	1			1		1			10
1967	2		1			2	8	3	3		4		23
1968	2			4		4	2	3	2				17
1969	8		2	3	1	2	4		2				22
1970	4			7		2	32	2	2				49
1971	4			2		8	4	2					20
1972	2			4		2	2		4		1		15
1973	2	1	1	2	1	6	4	2	2				21
1974	2	1		6		6	2	3				2	22
1975	2		1	4	1	4	2		2	1		2	19
1976	4		3	7			2	4	2			2	24
1977				4	1	2	2	2	2			2	15
1978		2		3	1	4	4		2			2	18
1979				1		2			2				5
1980				2		6	4	1	1				14

Table A-1 (Cont).

	Origin												
Year	EB	SAPS	KSO	MINS	PHNS	PSNS	NNS	PNS	CNS	WSO	NOR	INGL	TOTAL
1981					1		4		3				8
1982					1		6		3				10
1983		3		2		6	4		2	1			18
1984		7			1	6	4	2					20
1985						2	2	2	2				8
1986				2	1	4	4	2	2				15
1987				1		4		2	6				13
1988				4	1	5		3	4				17
1989				4	1	7		2	4				18
1990			3	4		10	4	4	3				28
1991				4	2	4		1	7				18
1992			3	3	2	7			4		4		23
1993					2	8	12						22
1994(1)			2	4		1	5		4				16
1995(1)				2		1							3
TOTAL	48	23	21	84	20	115	150	43	72	3	10	10	599

EB = Electric Boat Division of General Dynamics
SAPS = Shippingport Atomic Power Station
KSO = Kenneth A. Kesselring Site Operations
MINS = Mare Island Naval Shipyard
PHNS = Pearl Harbor Naval Shipyard
PSNS = Puget Sound Naval Shipyard
NNS = Newport News Shipbuilding
PNS = Portsmouth Naval Shipyard
CNS = Charleston Naval Shipyard
WSO = Windsor Site Operations
NOR = Norfolk Naval Shipyard
INGL = Ingalls Shipbuilding

(1) Shipments in these years cover those authorized by the court injunction.

6

plants are tested and qualified to characterize their performance for the lifetime of the plant. Part of this qualification program is to perform various irradiation tests of the materials for lifetime effects prior to certification. Along with those tests are pre- and post-examinations that provide the necessary data for subsequent analysis of the material in question. This work is considered a fundamental requirement for the design and safe operation of naval reactor plants. Therefore, the transfers of test specimen assemblies to the examination facility and shipments of the test specimens to the test facilities are included in the transportation evaluation. The test specimens have been assembled into test specimen assemblies and irradiated at the Test Reactor Area (TRA) on the INEL. The irradiated test specimen assemblies are returned to ECF for disassembly and examination.

Since 1956, approximately 3600 transfers of naval test specimen assemblies have been made between ECF and TRA in shipping casks transported by exclusive-use truck. For alternatives involving future transfers of this type, the transfers would be made in the NR-1, ATR-2, NR-3, NR-4, and Test Train casks. A detailed description of the shipping casks used to transfer irradiated test specimen assemblies is provided in Section A.4.3.

A.2.4 Shipments of Naval Irradiated Test Specimens to Examination and Testing Facilities

Following disassembly and examination of the test specimen assemblies at ECF, some specimens are shipped to off-site facilities for further testing or examination. These tests and examinations are generally very specialized and ECF does not have the capability to perform them or cannot perform them in a timely manner due to other examination priorities. Specimens are also shipped back to ECF for examination or further irradiation at TRA.

Test specimen shipments have been shipped to or from several laboratories and test facilities. They are the Bettis Atomic Power Laboratory (Bettis), located in West Mifflin, Pennsylvania; the Knolls Atomic Power Laboratory (KAPL), located in Niskayuna, New York; the Oak Ridge National Laboratory (ORNL), located in Oak Ridge, Tennessee; the Argonne National Laboratory (ANL)-East, located in Argonne, Illinois; the Battelle Memorial Institute, located in Columbus, Ohio; the Chalk River Nuclear Laboratories, located in Chalk River, Ontario, Canada (1 shipment only); the Hanford Site, located in Richland, Washington; and the ANL-West, Central Facilities Area (CFA), TRA, and ICPP facilities, all located on the INEL. Based on current schedules, Bettis and KAPL will be the

only origins for future shipments. Figure A-2 provides a map of the United States showing the transportation origins and destinations for the test specimen shipments.

Since 1956, approximately 850 shipments of naval test specimens have been made between ECF and on- and off-site testing and examination facilities, in shipping containers transported by exclusive-use truck. The shipments have been made in NRBK-41, -42, -43, and -44 shipping containers and the WAPD-39 and -40 shipping containers. For alternatives involving future shipments of this type, the shipments would be made in the NRBK-41 and WAPD-40 shipping containers. A detailed description of the shipping containers used to ship irradiated test specimens between off-site facilities and the examination facility is provided in Section A.4.4.

A.3 ALTERNATIVES TO BE EVALUATED

A detailed description of the alternatives is provided in Section 3. The specific impacts on each of the four types of naval shipments (described in Section A.2) are described below for each alternative.

A.3.1 Alternative 1 - No Action

Under this alternative, after implementation, there would be no further shipments of naval spent nuclear fuel from the shipyards and prototypes. The Expended Core Facility would be shut down. Naval spent nuclear fuel would be stored at a facility at the site where it was removed during reactor servicing, with the exception of naval spent nuclear fuel removed at Newport News Shipbuilding, a commercial shipyard, which would be transported to Norfolk Naval Shipyard for storage. All naval spent nuclear fuel currently at ECF would be transferred to ICPP prior to the start of the 40-year period with the exception of the fuel saved for future examinations, referred to as reference specimens. The reference specimens and the naval spent nuclear fuel which originated at the prototype sites at NRF would be shipped from ECF to ICPP sometime during the 40-year period. The TRA facility would perform any work associated with the assembly, disassembly, and routine examination of the test train assemblies; therefore, no transfers would be required. Specimens shipped off-site would remain at the destination following examination. Table A-2 summarizes the shipments for the No Action alternative.

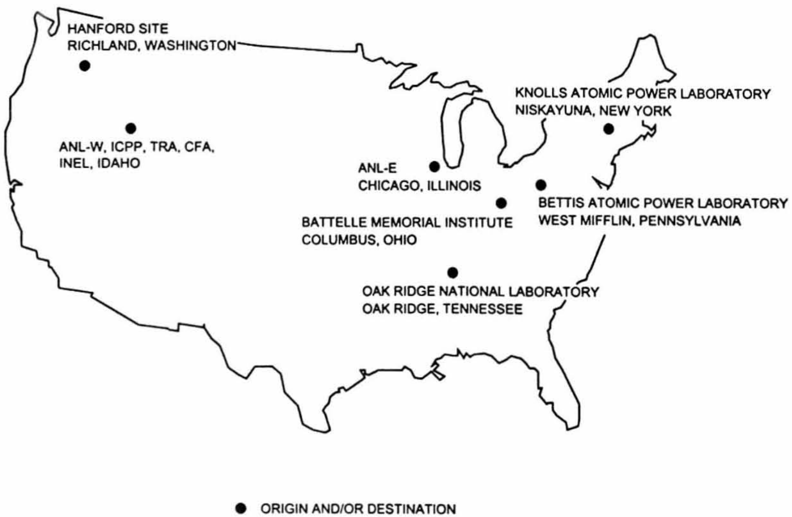

Figure A-2. Transportation origins and destinations for test specimen shipments.

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Table A-2. Summary of shipments for the No Action alternative.

Type of Shipment	
Shipments of Naval Spent Nuclear Fuel from Shipyards and Prototypes	
- Shipyards and Prototypes to ECF	None
- Newport News to Norfolk	Yes
Transfers of Naval Spent Nuclear Fuel from ECF to ICPP	Reference Specimens and Prototype Only
Transfers of Naval Test Specimen Assemblies Between ECF and TRA	None
Shipments of Irradiated Test Specimens Between Off-Site Facilities and ECF	
- Shipments from ECF	Yes
- Shipments back to ECF	None

A.3.2 Alternative 2 - Decentralization

As described in Section 3.4, this alternative also involves storage of the naval spent nuclear fuel near the point of origin. An evaluation of each of the three subalternatives defined in Section 3 was performed. The impact of the transportation related to each subalternative is briefly described below.

A.3.2.1 Alternative 2a - Store Naval Spent Nuclear Fuel at or Close to Locations Where Removed Without Examination. From the standpoint of transportation, this subalternative is equivalent to the No Action alternative.

A.3.2.2 Alternative 2b - Examine a Limited Amount of Naval Fuel in the Puget Sound Naval Shipyard Water Pit Facility and Store All Naval Fuel at Navy Facilities. For this alternative, the Expended Core Facility at NRF would be shut down and only high priority spent nuclear fuel would be transported to the Puget Sound Naval Shipyard for examination. For the naval spent nuclear fuel, approximately 10 percent of the total spent nuclear fuel for the 40-year period would be shipped. Following examination, the fuel would remain at Puget Sound Naval Shipyard. As in the No Action alternative, only the reference specimens would remain at ECF after June 1995. Ten percent of the reference specimens would be transferred from ECF to Puget Sound Naval Shipyard. The remainder of the reference specimens and the naval spent nuclear fuel which originated at the prototype sites at NRF would be transferred to ICPP. The TRA facility would perform any work associated with the assembly, disassembly, and routine examination of the test specimen assemblies; therefore, no transfers would be required. Shipments of test specimens to off-site facilities for specialized examinations would continue. Test specimens shipped off-site would remain at the destination following examination. Table A-3 summarizes the shipments.

Table A-3. Summary of shipments for the Decentralization - Limited Inspection alternative.

Type of Shipment	
Shipments of Naval Spent Nuclear Fuel from Shipyards and Prototypes	
- Shipyards and Prototypes to Puget Sound	Approximately 10% of spent fuel
- Newport News to Norfolk	Yes
Transfers of Naval Spent Nuclear Fuel from ECF to ICPP	Reference Specimens and Prototype Only
Transfers of Naval Test Specimen Assemblies Between Puget Sound and TRA	None
Shipments of Irradiated Test Specimens to Off-Site Facilities	
- Shipments from TRA	Yes
- Shipments back to TRA	None

A.3.2.3 Alternative 2c - Examine All Naval Spent Nuclear Fuel at the INEL and Return to Navy Facilities for Storage. For this alternative, all naval spent nuclear fuel would be shipped to ECF and examined as it has been in the past. Only non-destructive examinations would be performed. The spent nuclear fuel would be returned in the same condition as originally shipped. Following examination, the fuel would be returned to the originating shipyard or prototype site for storage in the same type of container with the exception that naval spent nuclear fuel which originated at Newport News Shipbuilding would be shipped to Norfolk Naval Shipyard for storage. New equipment would have to be designed and procured to handle the spent nuclear fuel which returns to the shipyard. As in the No Action alternative, only reference specimens would remain at ECF after June 1995. The naval spent nuclear fuel which originated in the prototype sites at NRF (A1W and S5G) would be transferred to ICPP. Transfers of the irradiated test specimen assemblies would continue, along with the shipments of test specimens from ECF to off-site testing or examination facilities. Specimens shipped off-site would remain at the destination following examination. Table A-4 summarizes the planned shipments for this alternative.

Table A-4. Summary of shipments for the Decentralization - Full Examination alternative.

Type of Shipment	
Shipments of Naval Spent Nuclear Fuel from Shipyards and Prototypes	
- Shipyards and Prototypes to ECF	Yes
- Newport News to Norfolk	To Norfolk from ECF
Transfers of Naval Spent Nuclear Fuel from ECF to ICPP	NRF Prototypes
Transfers of Naval Test Specimen Assemblies Between ECF and TRA	Yes
Shipments of Irradiated Test Specimens to Off-Site Facilities	
- Shipments from ECF	Yes
- Shipments back to ECF	None

A.3.3 Alternative 3 - 1992/1993 Planning Basis

This alternative plans on making the same types of shipments described in Section A.2 of this attachment. The only difference is that some of the historical origins of naval spent nuclear fuel and some destinations for the test specimen shipments will not be used. Table A-5 summarizes the planned shipments for this alternative.

Table A-5. Summary of shipments for the 1992/1993 Planning Basis alternative.

Type of Shipment	
Shipments of Naval Spent Nuclear Fuel from Shipyards and Prototypes	
- Shipyards and Prototypes to ECF	Yes
- Newport News to Norfolk	No
Transfers of Naval Spent Nuclear Fuel from ECF to ICPP	Yes
Transfers of Naval Test Specimen Assemblies Between ECF and ATR	Yes
Shipments of Irradiated Test Specimens to Off-Site Facilities	
- Shipments from ECF	Yes
- Shipments back to ECF	Yes

A.3.4 Alternative 4 - Regionalization

As described in Section 3.4, this alternative would distribute existing and new spent nuclear fuel between various sites either on the basis of the fuel type or on the basis of dividing storage between the eastern and western parts of the United States. An evaluation of each of the options for this alternative described in Section 3.4 was performed. The impact of the transportation related to each option under this alternative is briefly described below.

A.3.4.1 Alternative 4a - Regionalization Using Storage at Three Sites. From the standpoint of transportation of naval spent nuclear fuel and test specimens, this alternative is equivalent to the 1992/1993 Planning Basis alternative.

A.3.4.2 Alternative 4b - Regionalization Using Storage at Two Sites. This alternative would utilize an existing DOE site in the eastern part of the United States and another existing DOE site in the western part of the country for storage of spent nuclear fuel. From the standpoint of transportation of naval spent nuclear fuel and test specimens, this alternative is equivalent to the Centralization alternative at each of the DOE sites because the Navy would operate a facility for examining naval spent nuclear fuel at only one of the DOE sites and the naval spent nuclear fuel would be stored at the same site where it was examined.

A.3.5 Alternative 5 - Centralization

This alternative considers consolidating all naval spent nuclear fuel and test specimens at the INEL, Hanford Site, Savannah River Site, Oak Ridge Reservation, or Nevada Test Site. Centralization at INEL is identical to the 1992/1993 Planning Basis alternative. For the other centralization sites, the type and number of shipments would be identical to the 1992/1993 Planning Basis alternative with the only difference being the destination. The naval spent nuclear fuel will be shipped to the centralization site for examination and subsequently transferred to a storage facility at the centralization site which would be equivalent to ICPP. Naval spent nuclear fuel shipments from Newport News Shipbuilding to Norfolk Naval Shipyard would not be necessary. As in the No Action alternative, only reference specimens would remain at ECF after June 1995. All reference specimens would be shipped to the centralization site. The naval spent nuclear fuel which originated in the prototype sites at NRF would also be transferred to the centralization site. The test specimen

assembly shipments would be shipped between TRA and the alternate site. The test specimen shipments would originate at the centralization site and all specimens would ultimately return to that site for storage.

A.4 GENERAL DESCRIPTIONS

The following general information is common to all of the alternatives evaluated.

A.4.1 Spent Nuclear Fuel Shipping Containers

For naval spent nuclear fuel, the M-130, M-140, and M-160 shipping containers would be used for all alternatives. The shipping containers are primarily transported by railcars used only for this purpose as part of general-use freight trains. Section A.2.1 describes the special circumstances where the shipping containers are transported by ship or heavy-lift transporter. A brief description of each shipping container follows.

A.4.1.1 M-130 Shipping Container. The M-130 shipping container is a large, lead-lined, steel-shelled shipping container that is transported in the vertical position on a depressed center railcar (Figure A-3). The major components of the M-130 shipping container include the shielded container, closure head, and dust cover. Module holders are installed inside the container to hold the irradiated fuel modules in place and can be modified to accept different sized fuel modules. The container is shipped dry with the exception of a small amount of residual water. Cooling fins on the outside of the container are designed to dissipate the heat generated by the spent nuclear fuel.

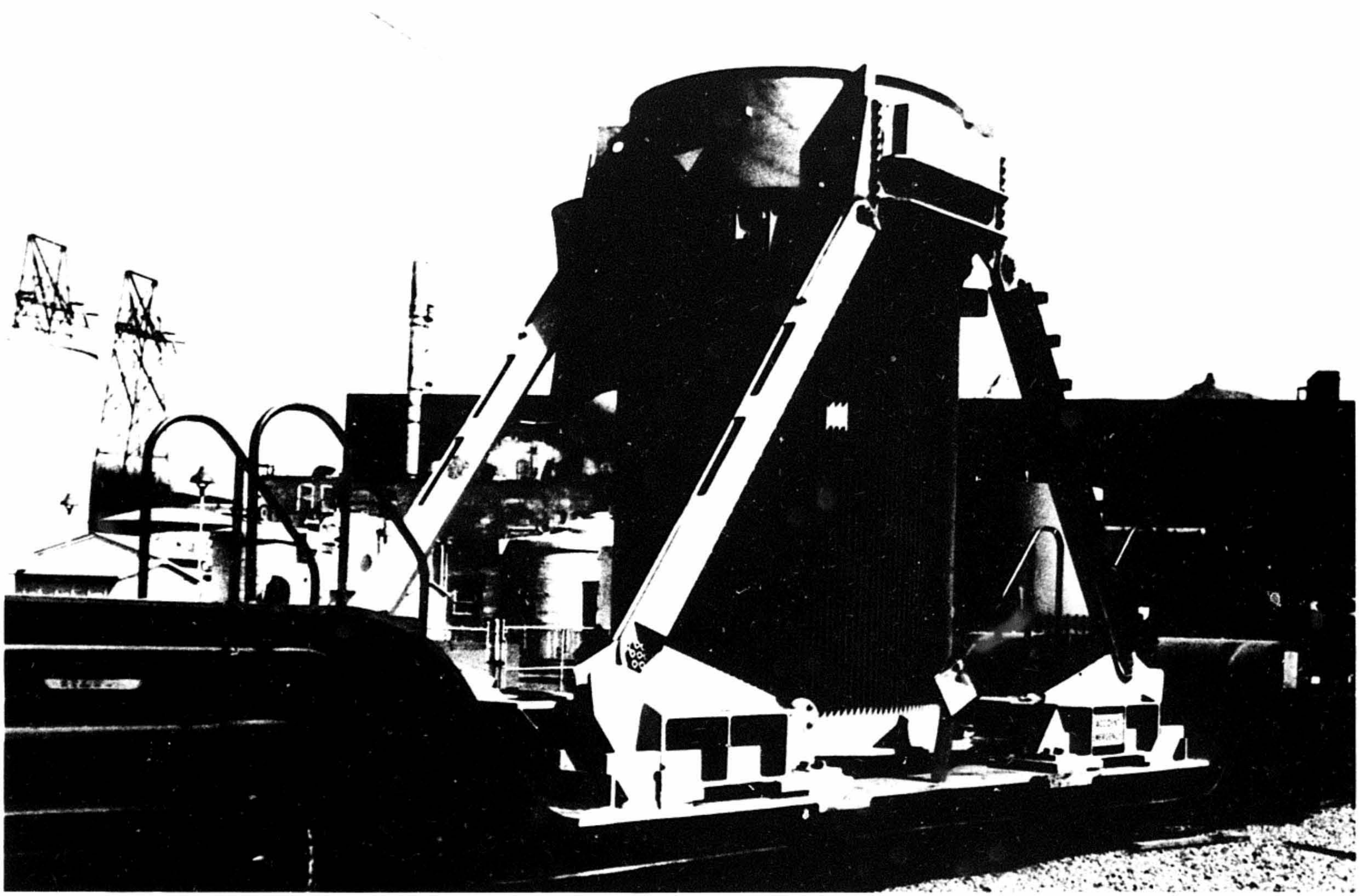

Figure A-3. M-130 shipping container mounted on railcar.

The M-130 shipping container weighs approximately 214,500 pounds in the standard loaded configuration. The container is approximately 13 feet tall and 7 feet in diameter. The container is a closed bottom cylindrical lead shell that is covered both on the inside and the outside with a 1-inch thick layer of steel. The lead on the cylindrical sides is about 10 inches thick and is a minimum of 9.5 inches thick on the bottom. In the standard configuration, the closure head at the top of the container is primarily constructed of 5.25 inches of lead and 7 inches of steel.

A.4.1.2 M-140 Shipping Container. The M-140 shipping container is a large, stainless steel shipping container that is transported in the vertical position on a specially designed well-type railcar (Figure A-4). The major components of the M-140 shipping container include the shielded container, closure head, and protective dome. Module holders are installed inside the container to hold the irradiated fuel modules in place and can be modified to accept different sized fuel modules. The container is shipped dry with the exception of a small amount of residual water. Cooling fins on the outside of the container are designed to dissipate the heat generated by the fuel.

The M-140 shipping container weighs approximately 375,000 pounds in the loaded condition. The container is approximately 16 feet tall with a maximum diameter of 10.5 feet. The container body is made from stainless steel forgings with 14-inch thick walls and a 12-inch thick bottom. The closure head and protective dome have a total thickness of 17.5 inches of stainless steel.

A.4.1.3 M-160 Shipping Container. The M-160 shipping container is a large, lead-lined, steel-shelled shipping container that is transported in a horizontal position on a support structure mounted on a modified flat bed railcar (Figure A-5). The major components of the M-160 shipping container include the shielded container, closure head, and dust cover. Module holders are installed inside the container to hold the irradiated fuel modules in place. The container is shipped dry with the exception of a small amount of residual water. Cooling fins on the outside of the container are designed to dissipate the heat generated by the fuel.

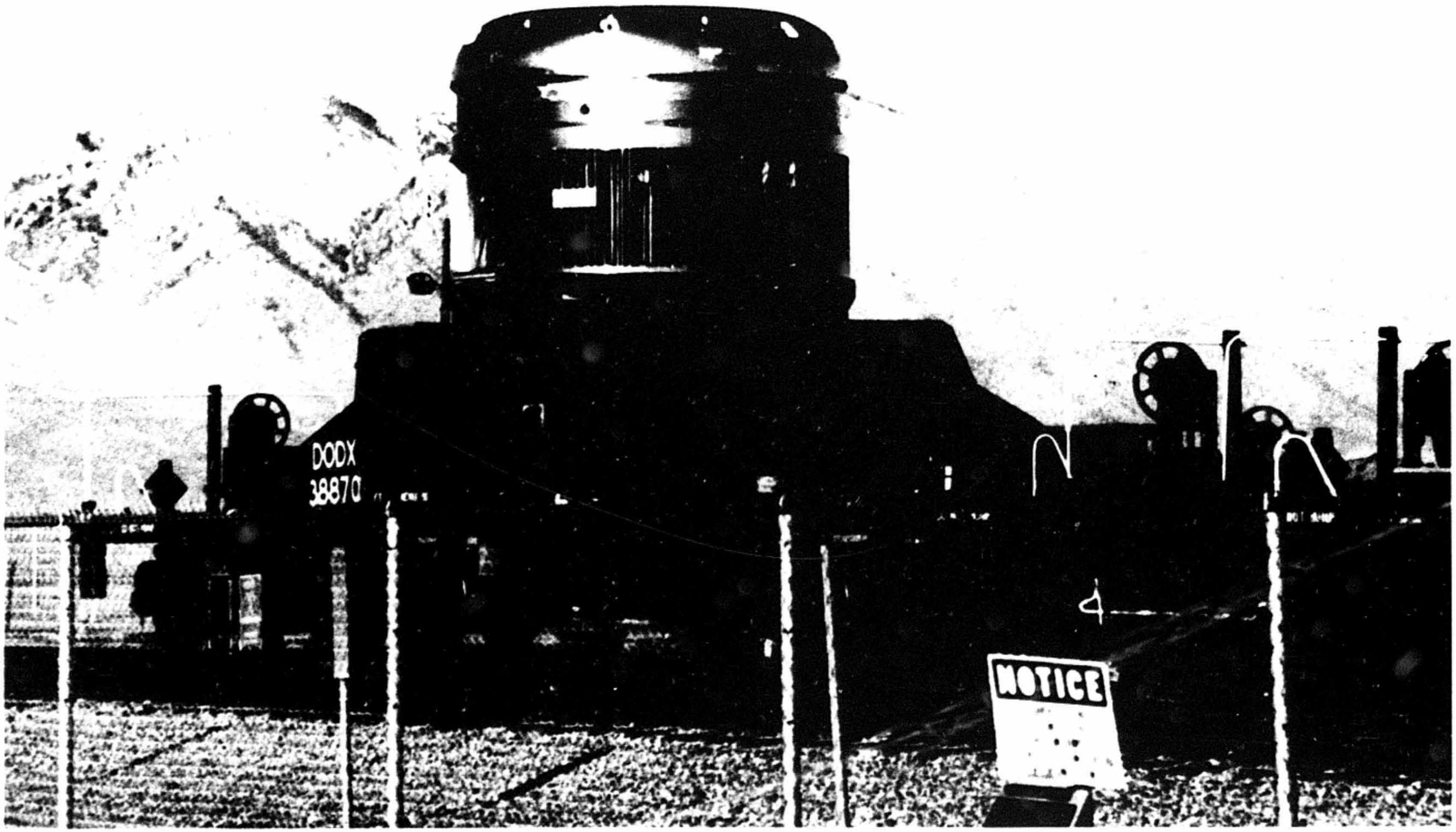

Figure A-4. M-140 shipping container mounted on railcar.

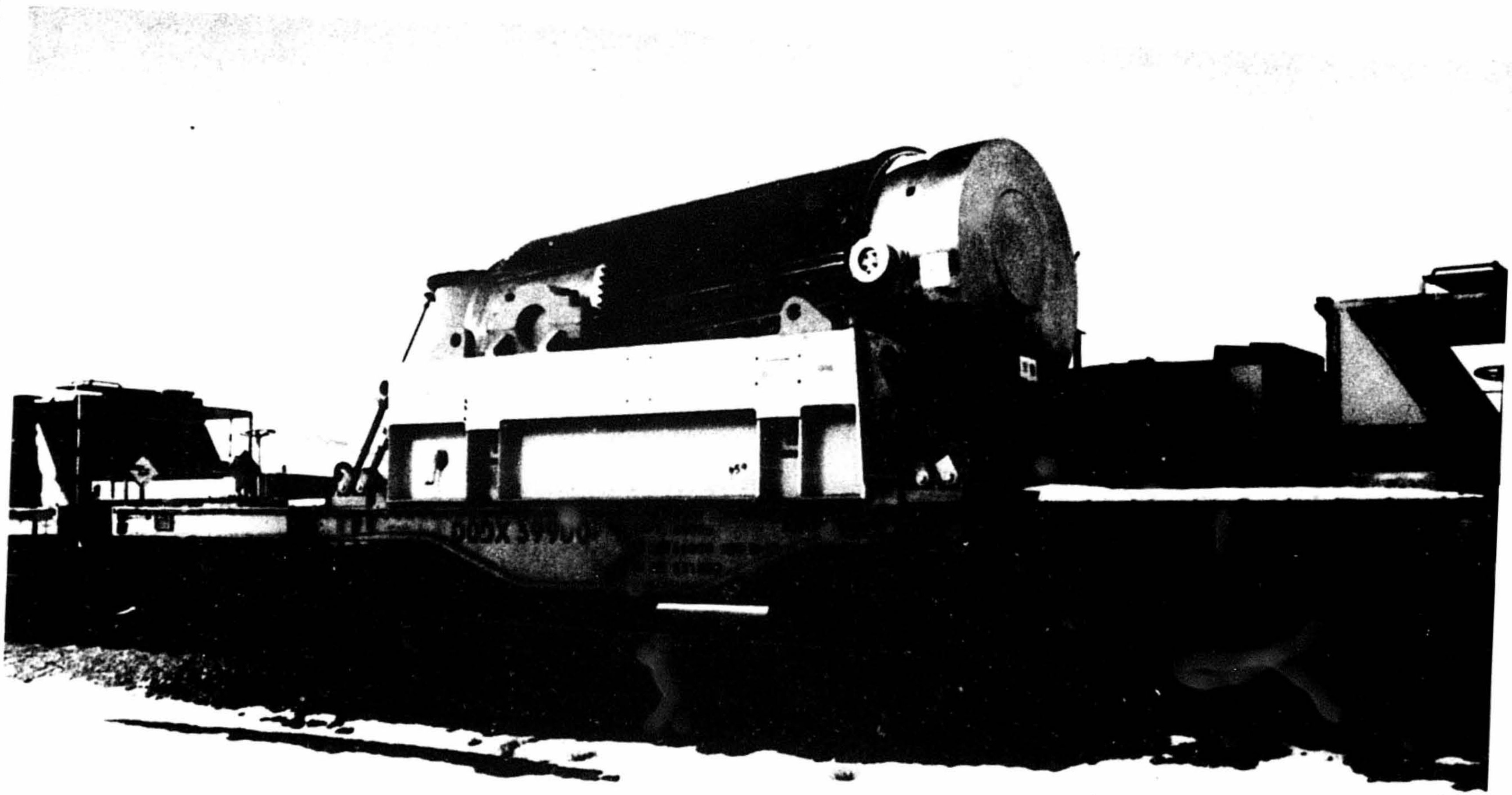

Figure A-5. M-160 shipping container mounted on railcar.

18

The M-160 shipping container weighs approximately 235,500 pounds in the loaded condition. The container is approximately 16.5 feet long and 6.5 feet in diameter. The container consists of two concentric bottom closed steel cylinders with a 9.4-inch annulus between the cylinders that is filled with lead. The outer shell is made from 1.5-inch thick steel, and the inner shell is made from 1-inch thick steel. The bottom plate is approximately 7 inches thick, and the closure head is approximately 15 inches thick.

A.4.1.4 Government Escorts for Spent Nuclear Fuel. Commercial railroads, exclusive-use heavy-lift transporters, or exclusive-use ships are used to transport the naval spent nuclear fuel from the prototypes and shipyards. The specific routes used to transport the spent nuclear fuel are selected by the rail or shipping companies. All naval spent nuclear fuel shipments are accompanied by government escorts. The escorts perform the duties necessary to ensure the safe, expeditious transportation of the naval spent nuclear fuel.

The government escorts receive specialized training in shipment safety procedures, radiological controls, security, and emergency response. Routine shipment escort procedures involve processing of authorization and shipping documentation, pre-shipment inspections, tracking shipment progress and schedules, enroute inspections, shipment observation and surveillance, and periodic communication checks. The government escorts have been trained to use and are equipped with the necessary radiological monitoring equipment to verify the shipping container integrity.

A large amount of the government escorts' training involves emergency response. This training involves emergency procedures for notification of technical and safeguards support personnel. The government escorts are equipped to immediately notify emergency assistance personnel, immediately assess the containment status of the shipping container, and communicate this information to emergency support personnel. Depending on the situation, the technical and support personnel may activate various emergency control centers that are prepared to provide the government escorts with the necessary support to quickly and safely bring an emergency situation under control. All railroads, which handle escorted shipments, also have specific emergency response procedures to safely expedite recovery for shipments that are involved in a rail line accident. Continually manned railroad operation centers maintain the capability to contact personnel from a combination of resources which provide appropriate equipment and manpower at the accident scene.

A.4.2 Spent Nuclear Fuel Shipping Casks for Transfers to Storage Following Examination

For naval spent nuclear fuel being transferred from the examination facility to storage (e.g., ECF to ICPP), the Nuclear Fuel Services Model 100 cask (NFS-100), Peach Bottom cask, and the Large Cell cask will be used for all alternatives. These shipping containers are transported by exclusive-use truck. A brief description of each cask follows.

A.4.2.1 NFS-100 Cask. The NFS-100 cask is a large, lead-lined, steel-shelled shipping cask that is transported in the horizontal position on a skid assembly attached to a tandem axle trailer (Figure A-6). The major components of the NFS-100 cask include the shielded cask and closure head. A fuel holding insert is installed inside the cask to hold the irradiated fuel modules in place. The container is shipped dry with the exception of a small amount of residual water. The cask is enclosed on the truck by a metal cover during shipment.

The NFS-100 cask weighs approximately 110,000 pounds in the loaded configuration. The cask is approximately 10.5 feet tall and 7 feet in diameter. The cask is a closed bottom cylinder of lead with a 0.375-inch thick steel inner shell and a 2-inch thick outer shell. The lead on the cylindrical sides is about 8.75 inches thick and the lead on the bottom is 8.8 inches thick. The closure head at the top of the cask is constructed of 9.75 inches of lead and 2 inches of steel.

A.4.2.2 Peach Bottom Cask. The Peach Bottom cask is a large, lead-lined, steel-shelled shipping cask that is transported in the horizontal position on a skid assembly attached to a tandem axle trailer (Figure A-7). The major components of the Peach Bottom cask include the shielded cask and closure heads. A fuel holding insert is installed inside the cask to hold the irradiated fuel modules in place. The cask is shipped dry with the exception of a small amount of residual water. The cask is enclosed on the truck by a metal cover during shipment.

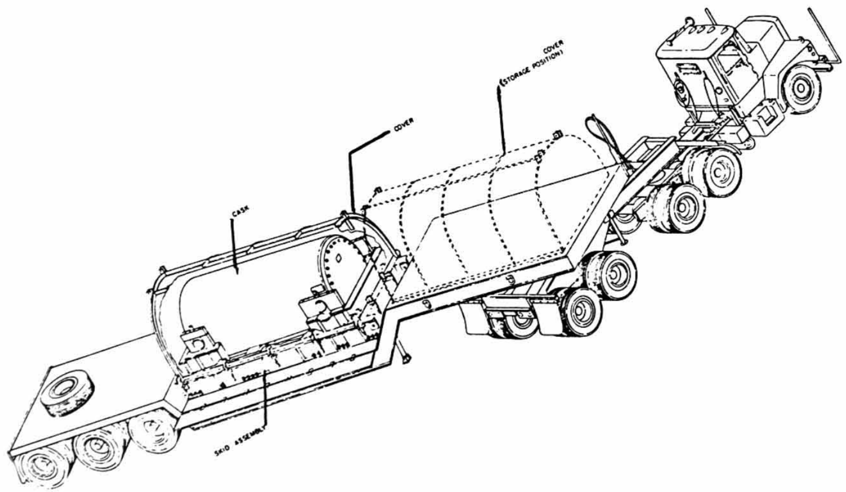

Figure A-6. NFS-100 cask mounted on truck.

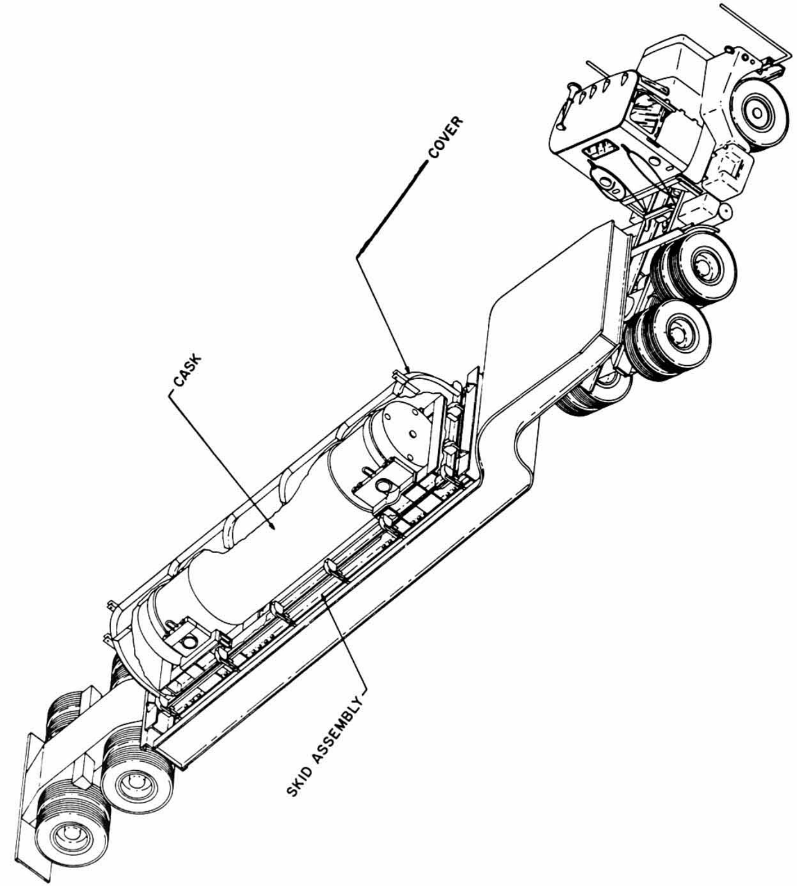

Figure A-7. Peach Bottom cask mounted on truck.

The Peach Bottom cask weighs approximately 68,400 pounds in the loaded configuration. The cask is approximately 16 feet tall and 3.5 feet in diameter. The cask is a stepped cylinder of lead with a 0.25-inch thick steel inner shell and a 1.75-inch thick steel outer shell. The lead on the cylindrical sides ranges from 5.25 to 6.25 inches thick. The closure heads on each end of the cask are essentially identical and are constructed of 8.5 inches of steel.

A.4.2.3 Large Cell Cask. The Large Cell cask, currently being designed for larger fuel types, will be a large, stainless steel shipping cask that is transported in the vertical position on a low-boy tractor trailer (Figure A-8). The major components of the Large Cell cask will include a shielded cask, closure head, shipping cask, and external impact limiters. Fuel-holding inserts will be installed inside the cask to hold the irradiated fuel modules in place. The cask will be shipped dry with the exception of a small amount of residual water. Cooling fins on the outside of the shipping cask are designed to dissipate the heat generated by the fuel.

The Large Cell cask will weigh approximately 220,000 pounds in the loaded condition. The shielded cask will be approximately 14 feet tall and 7 feet in diameter. The shielded cask body will be a closed bottom cylinder made from stainless steel forgings with 13.5-inch thick walls and a 13-inch thick bottom. The closure head will be a 14-inch thick stainless steel forging. The shielded cask will be assembled to the shipping cask during transport. The shipping cask will be a 2-inch thick aluminum closed bottom cylinder with fins extending to a total diameter of 93.6 inches. The external impact limiter assemblies, located on both ends of the cask, will be constructed of encased bi-directional aluminum honeycomb and are approximately 10 feet in diameter. The total Large Cell cask height will be approximately 17 feet.

A.4.2.4 Shipment Controls. All spent nuclear fuel transfers to a storage facility at the same site as the examination facility will be accompanied by escorts. The escorts are personnel who are specially trained to perform the duties necessary to ensure the safe transportation of the spent nuclear fuel. The escorts are in vehicles located in front of and behind the truck carrying the shipping cask.

The escorts receive specialized training in shipment safety procedures, radiological controls, security, and emergency response. The escort vehicles are equipped with distinctive warning flashers, and the escorts are capable of radio contact with each other, the driver of the transport vehicle, and on-site emergency coordinating personnel.

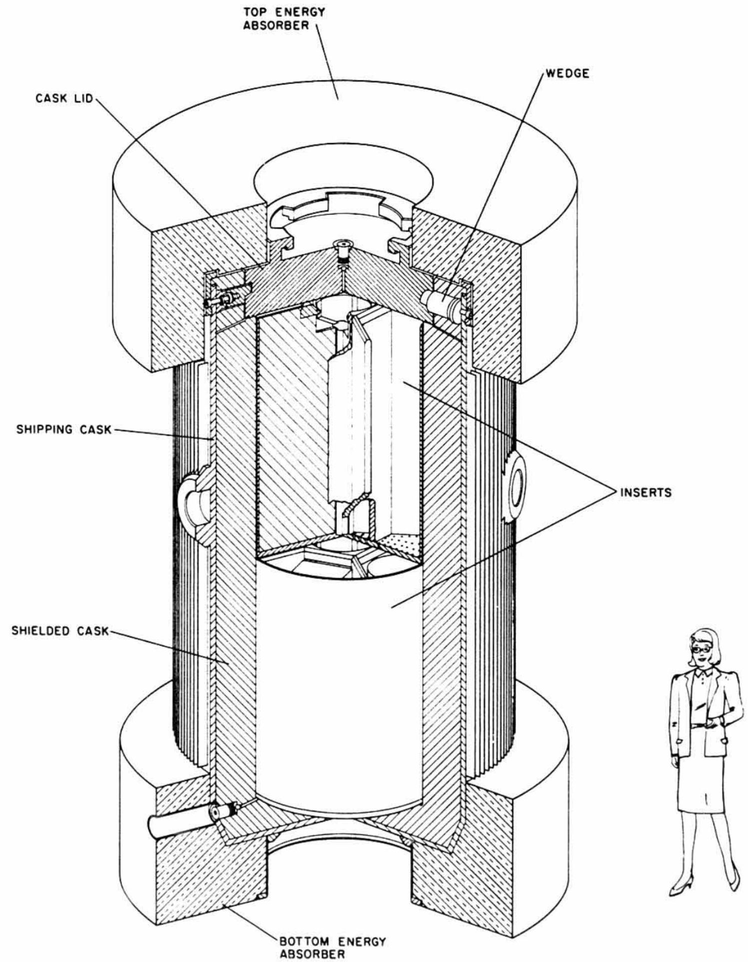

Figure A-8. Large Cell cask.

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A large amount of the escorts' training involves emergency response. This training involves emergency procedures for notification of site technical and safeguards support personnel. The escorts are equipped to immediately notify emergency assistance personnel, immediately assess the containment status of the shipping cask, and communicate this information to emergency support personnel. Depending on the situation, the technical and support personnel may activate various emergency control centers that are equipped with the equipment and manpower to provide the escorts with the necessary support to quickly and safely bring an emergency situation under control.

Additional administrative controls are imposed on the transfers to further minimize risks. For example, the transfers are not allowed to travel during heavy traffic periods such as shift changes, and the convoy travels at reduced speeds. The route itself also enhances safety, since the route is essentially flat and the highest possible drop distance in the event of an accident is approximately 5 meters (16.5 feet) at the location where the highway crosses a river bed.

A.4.3 Naval Test Specimen Assembly Casks for Transfers Between TRA and the Examination Facility

For naval test specimen assemblies being transferred on-site between TRA and the examination facility, the NR-1, ATR-2, NR-3, NR-4, and Test Train casks will be used. These casks are transported by exclusive-use truck. For off-site shipments to the examination facility at the centralization sites, only the Test Train cask will be used. A brief description of each cask follows.

A.4.3.1 NR and ATR Casks. The NR and ATR casks are large, lead-lined, steel-shelled casks that are transported approximately 10° off horizontal in a cradle assembly attached to a tandem trailer (see Figure A-9). The major components of the casks include the shielded body, mast, and bottom closure/shield.

The shielded bodies of the casks are all approximately 32 inches in diameter. The outer steel shell thickness ranges from 0.5 inch to 1.0 inch. The thickness of the inner steel shell is approximately 0.4 inch for each cask. The lead ranges from approximately 10 inches to 11 inches for the various casks. The height of the shielded body ranges from approximately 6 feet to 12 feet. The mast is a tower section formed of reinforced aluminum and serves to support the structural end of the

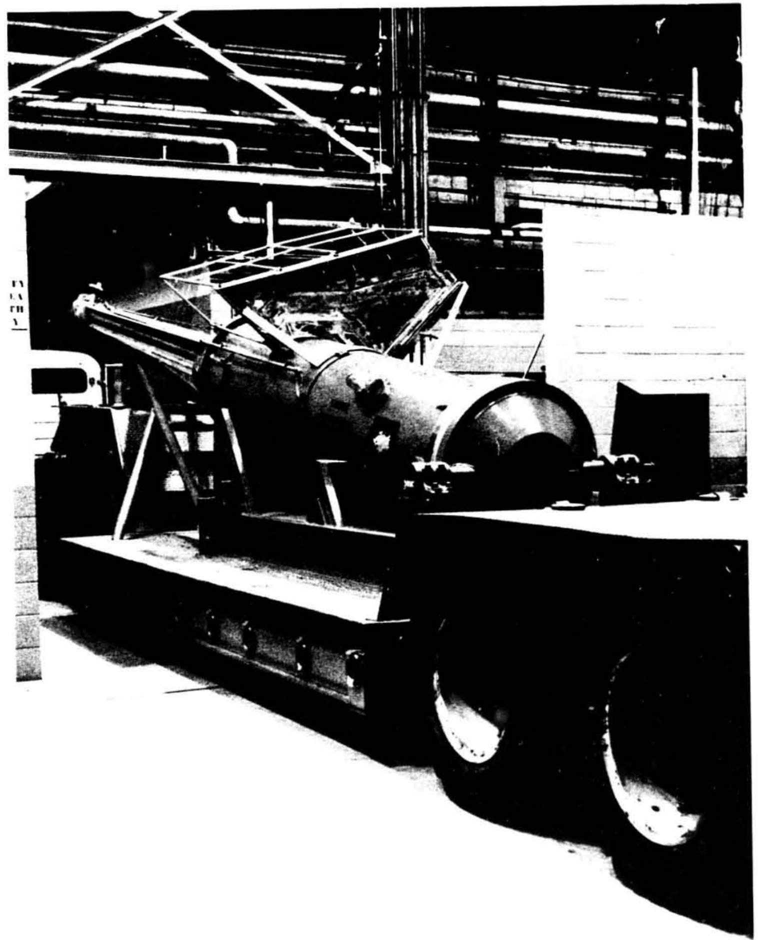

Figure A-9. NR/ATR cask mounted on truck.

specimen assemblies which require very little shielding. A winch and platform are also attached to each cask. The bottom closure/shield is constructed of 1.0 to 1.75 inches of steel and 7.0 to 8.75 inches of lead.

The NR and ATR casks range in weight from approximately 19,000 to 48,000 pounds. The overall cask height ranges from approximately 20 to 30 feet.

A.4.3.2 Test Train Casks. A new test specimen container would be required to transport irradiated test specimen assemblies between TRA and the examination facility located at the sites other than INEL for the Centralization alternative. A new cask is currently being designed to replace the current casks used to transport the test specimen assemblies between ECF and TRA, which are approaching the end of their design lifetime. The basic concept for this new cask is a thick-walled, stainless steel body with stainless steel closures on each end. Energy absorbers will be attached to the cask to prevent damage to the test specimens. The current estimated size of this cask is 34 feet long by 5 feet in diameter, weighing approximately 40 tons. This cask would be shipped by exclusive-use truck.

A.4.3.3 Shipment Controls. All spent nuclear fuel transfers to an examination facility at the same site as the irradiation facility will be accompanied by two escorts. The escorts are personnel who are specially trained to perform the duties necessary to ensure the safe transportation of the spent nuclear fuel. The escorts are in vehicles located in front of and behind the truck carrying the shipping cask.

The escorts receive specialized training in shipment safety procedures, radiological controls, security, and emergency response. A large amount of the escorts' training involves emergency response. This training involves emergency procedures for notification of site technical and safeguards support personnel. The escorts are equipped to immediately notify emergency assistance personnel, immediately assess the containment status of the shipping cask, and communicate this information to emergency support personnel. Depending on the situation, the technical and support personnel may activate various emergency control centers that are equipped with the equipment and manpower to provide the escorts with the necessary support to quickly and safely bring an emergency situation under control. The escort vehicles are equipped with distinctive warning flashers, and the escorts are capable of radio contact with each other, the driver of the transport vehicle, and emergency coordinating personnel.

Additional administrative controls are imposed on the shipments to further minimize risk. For example, the transfers are not allowed to travel during heavy traffic periods such as shift changes, and the convoy travels at reduced speeds. The route itself also enhances safety, since the route is essentially flat and the maximum possible drop in the event of an accident is from the bed of the truck to the road bed.

For the Centralization alternative, the casks would be shipped off-site. In this instance, only casks certified for over-the-road transportation in accordance with the Nuclear Regulatory Commission regulations would be used for shipments of the test trains. No escorts or additional administrative controls would be used.

A.4.4 Test Specimen Shipping Containers

For test specimens, the WAPD-40 and NRBK-41 shipping containers would be used to transport the specimens between ECF and the off-site laboratories and test facilities for all alternatives. These shipping containers are transported by an enclosed truck using a commercial carrier. A brief description of each container follows.

A.4.4.1 WAPD-40 Shipping Container. The WAPD-40 shipping container (Figure A-10) is a cylindrical, lead-shielded, steel-clad container that is shipped in a horizontal position. The inner steel shell is 0.25-inch thick, and the outer steel shell is 0.5-inch thick with 9.875 inches of lead shielding in between. The container is approximately 13 feet long and 2 feet in diameter. Steel clad, lead-shielded end plugs bolt onto each end, and 0.5-inch thick plates are bolted over the end plugs. The specimens are placed into special sealed inner containers prior to placement into the WAPD-40 shipping container. The weight of the container and skid assembly is approximately 28,000 pounds. The container and skid assembly are mounted into a special holddown cradle on the truck. This holddown cradle weighs approximately 5,000 pounds.

A.4.4.2 NRBK-41 Shipping Container. The NRBK-41 shipping container (Figure A-11) is a cylindrical, lead-shielded, steel-clad container that is shipped in the vertical position. The inner steel shell is 0.25-inch thick, and the outer steel shell is 0.5-inch thick with 10 inches of lead shielding in between. The container has a 1-inch thick steel plate welded to the bottom with a second 1-inch thick steel plate welded to the first plate with a 0.125-inch deep recess to provide a thermal break for the

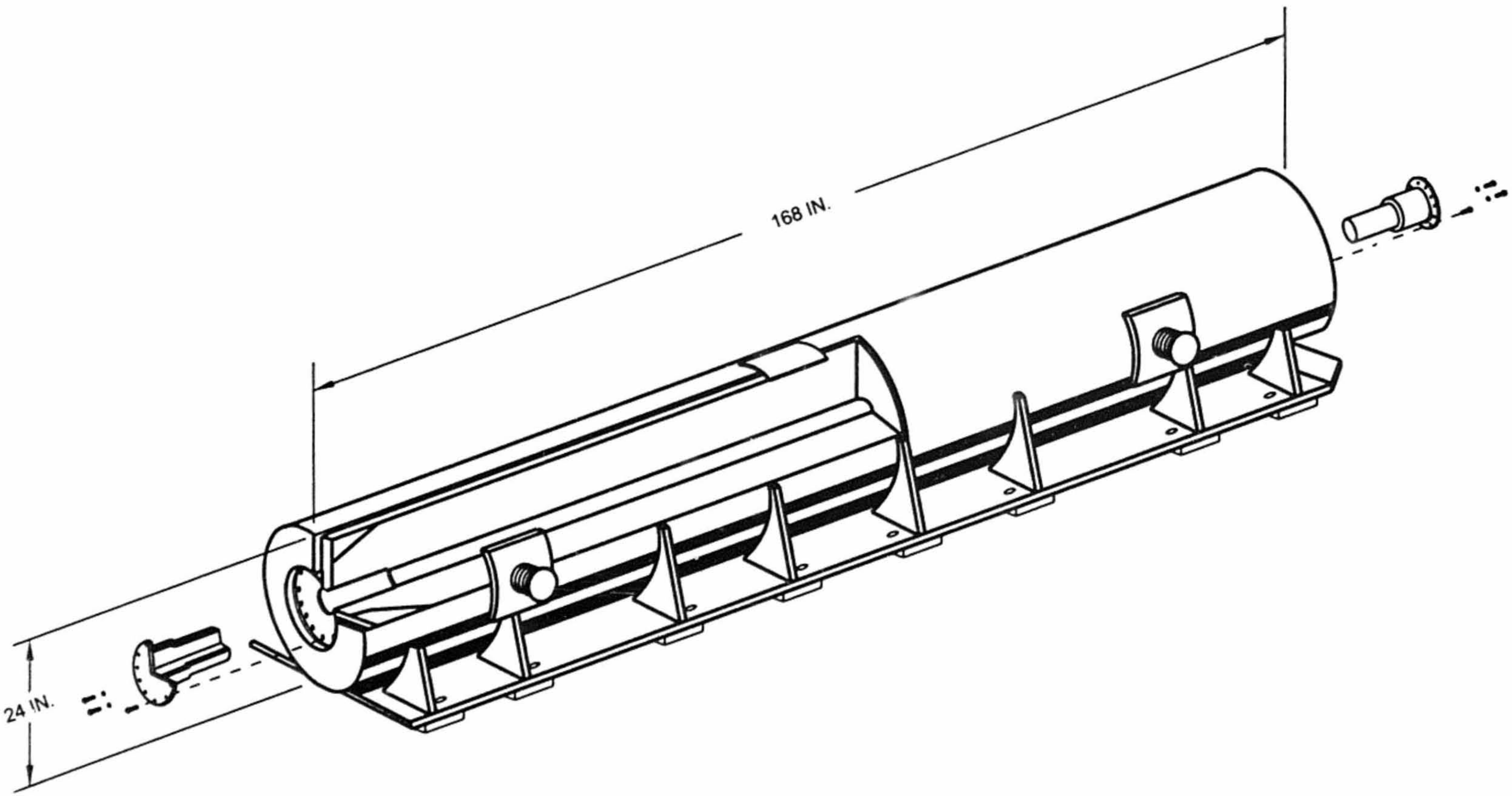

Figure A-10. WAPD-40 shipping container.

29

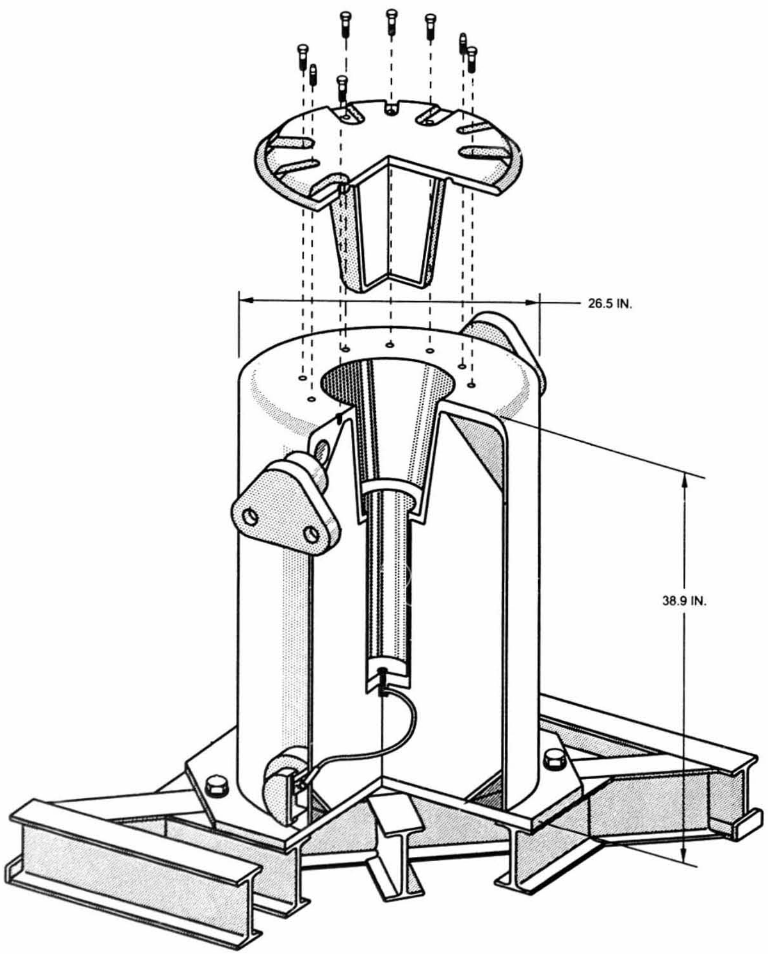

Figure A-11. NRBK-41 shipping container.

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bottom of the container. The container also has a 0.25-inch thick steel outer thermal shield attached that provides a 0.125-inch air gap between the outer shell and the thermal shield. The container is approximately 4 feet tall and 2.25 feet in diameter. The container is bolted to a welded 48-inch square I-beam skid that is used to distribute the container load. The specimens are placed into a special sealed inner container prior to placement into the NRBK-41 shipping container. The weight of the loaded container is approximately 9,000 pounds.

A.4.5 Shipping Container Design Requirements

The M-130, M-140, M-160, NRBK-41, and WAPD-40 shipping containers have been designed and built to meet the regulations specified in Title 49, Code of Federal Regulations, Part 173 (49CFR173), entitled " Shippers - General Requirements for Shipments and Packagings" (CFR 1991). Shipments of naval spent nuclear fuel and test specimens are further regulated by Title 10, Code of Federal Regulations, Part 71 (10CFR71), entitled "Packaging of Radioactive Material for Transportation and Transportation of Radioactive Material Under Certain Conditions" (CFR 1993). These regulations require the shipping container to meet specific criteria under normal transport and accident conditions. The shipping container must be evaluated under free drop, puncture, heat, cold, pressure, water spray, and vibration for normal conditions and a series of severe hypothetical accident conditions with the results compared against the criteria provided in 10CFR71.

The M-130, M-140, M-160, WAPD-40, and NRBK-41 shipping containers have undergone rigorous engineering evaluations to assure compliance with 49CFR173 and 10CFR71 requirements. In addition, actual scale model or mock-up tests have been performed to verify selected engineering evaluations. This compliance has been certified by the U. S. Department of Energy and the Nuclear Regulatory Commission. The new Test Train and Large Cell casks will also be designed in accordance with the requirements of 49CFR173 and 10CFR71 and will undergo the same rigorous engineering evaluations and testing.

The safety analyses for the NFS-100, Peach Bottom, NR, and ATR casks demonstrate compliance with the requirements specified by the Department of Energy (DOE) in DOE Order 5480.3, entitled "Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes" (DOE 1985) and supplemented by DOE Idaho Operations Office Order ID 5480.3, entitled "Hazardous Materials Packaging and Transportation

Safety Requirements" (DOE 1991). These requirements are similar to the requirements of 10CFR71 with the major difference being that a worst credible accident can be defined based on site-specific information.

The NFS-100, Peach Bottom, NR, and ATR casks have undergone rigorous engineering evaluations to assure compliance with the DOE requirements. In addition, actual scale model or mock-up tests have been performed to verify selected engineering evaluations. The shipping casks comply with the requirements of DOE 5480.3 and DOE ID 5480.3 and this compliance is demonstrated by approval from the Idaho Operations Office of the Department of Energy.

A.5 TECHNICAL APPROACH - GENERAL

Several computer codes were used to assess the radiological risks associated with the transportation of naval spent nuclear fuel and test specimens. Specifically, the RADTRAN 4 risk analysis model, developed by Sandia National Laboratories (Neuhauser and Kanipe 1992), was used to calculate the general population and transportation crew (occupational) radiological risks associated with the transportation of radioactive materials. This computer code was used extensively in the incident-free and accident risk assessments. In some cases, other methods were more appropriate than the RADTRAN 4 computer code for naval spent nuclear fuel. In these cases, other calculational models were used and are specifically identified.

The RISKIND computer code, developed by Argonne National Laboratory (Yuan et al. 1993), also specifically analyzes radiological consequences and health risks to individuals from exposure associated with transportation. For incident-free evaluations, RISKIND uses a generic truck cask and does not allow adjustments for different sized casks which is not appropriate for naval spent nuclear fuel and test specimen casks; therefore, this code was not used. RISKIND (a version which accepts fuel-specific isotopes) was found to be the best code for calculation of the maximum individual and general population consequences for the accident scenario and was used for that purpose.

Several other computer codes were used to provide input for the RADTRAN 4 and RISKIND computer codes. The codes include INTERLINE, HIGHWAY, SPAN4, and ORIGEN2. A description of each computer code and how the code was used is provided below.

The INTERLINE computer code, developed by Oak Ridge National Laboratory (Johnson et al. 1993a), was used to evaluate the rail routes used for the spent nuclear fuel shipments.

The HIGHWAY computer code, also developed by Oak Ridge National Laboratory (Johnson et al. 1993b), was used to evaluate the truck routes used for the test specimen shipments.

The SPAN4 computer code (Wallace 1972) was used to perform gamma exposure rate calculations for the various shipping containers to assess the effect of increased distance from the source on exposure. SPAN4 is a point kernel code where appropriate exponential kernels are integrated over a source distribution. SPAN4 was developed by the Bettis Atomic Power Laboratory specifically for naval spent nuclear fuel.

The ORIGEN2 is a computer code, developed by Oak Ridge National Laboratory (Croff 1980), that is used to simulate radiation and decay of materials that are irradiated in a nuclear reactor. The ORIGEN2 computer code is widely accepted in the public domain and was used to independently confirm the fission product inventory for naval fuel developed using the standard Bettis Atomic Power Laboratory method. In addition, the standard Bettis Atomic Power Laboratory method has been used in Safety Analysis Reports for Packaging, reviewed and accepted by the Nuclear Regulatory Commission.

The radiological risks associated with the transportation of spent nuclear fuel and irradiated test specimens have been assessed for the general population, transportation workers (occupational), and hypothetical maximum exposed individuals under incident-free and accident conditions for the alternatives presented in Section A.3. The maximum consequences for an accident are also provided for each alternative. The radiation exposure to the government escorts for shipments was considered occupational in nature and was included with the transportation worker results.

The radiological impacts are first expressed as the calculated total exposure for the exposed population, occupational workers, and the maximum exposed individuals. The calculated total exposures are then used to estimate the hypothetical health effects, expressed in terms of estimated cancer fatalities. The health risk conversion factors used in this evaluation are taken from the International Commission on Radiological Protection (ICRP Publication 60) which specifies 0.0005 fatal cancer cases per person-rem for members of the public, 0.0004 fatal cancer cases per person-rem for workers (ICRP 1991). To calculate the estimated health detriment, the calculated exposure would be multiplied by the conversion factors of 0.00073 health detriments per person-rem for members of the public, and 0.00056 health detriments per person-rem for workers (ICRP 1991).

The numerical estimates of cancer deaths and other health detriments presented were obtained by the practice of linear extrapolation from the nominal risk estimate for lifetime total cancer mortality at 10 rad. Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of cancer deaths. Studies of human populations exposed at low doses are inadequate to demonstrate the actual level of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (CIRRPC 1992). In this appendix, the doses have been provided in all cases to allow independent evaluation using any relation between exposure and health effects.

Non-radiological risks related to the transportation of naval spent nuclear fuel are also estimated. The non-radiological risks are associated with vehicle exhaust emission for incident-free transportation and fatalities resulting from transportation accidents. The non-radiological risks associated with shipments that return empty containers to the origin are also included. Risk factors for vehicle exhaust emissions and state-level accident fatality rates were obtained from "Non-Radiological Impacts of Transporting Radioactive Material" (Rao et al. 1982), "Transportation Impacts of the Commercial Radioactive Waste Management Program" (Cashwell et al. 1986), and "Longitudinal Review of State-Level Accident Statistics for Carriers of Interstate Freight" (Saricks and Kvitek 1994), respectively.

The shipments of radioactive waste at shipyards are not addressed. The exposure related to incident-free transportation would be small and would be the same for all alternatives which would not affect the decision-making process. The consequences of an accident would also be insignificant compared to the accidents analyzed for spent nuclear fuel.

For the ocean-going portion of the shipments of naval spent nuclear fuel from shipyards and prototypes, there would be no exposure to the general population. The basis for this conclusion is that the ship's hull provides a considerable amount of additional shielding and that there would be no members of the general population close enough to the ship to receive appreciable exposure during these shipments. The consequences of an accident during the ocean-going portion have also not been evaluated because the forces on the container during an accident aboard the ship would not be large

enough to cause damage to the container or fuel inside it since the ship itself would sustain the direct impact. This is substantiated by the fact that the impact forces to the container would be less than the regulatory criteria. Therefore, no release would occur.

A.5.1 Technical Approach for the Assessment of Incident-free Transportation

For incident-free transportation of naval spent nuclear fuel, the RADTRAN 4 computer code was used to calculate the radiological exposure for the general population and a portion of the occupational exposure.

Included in the RADTRAN 4 computer code incident-free risk calculations for transport are models describing (1) exposures to persons (e.g., residents) adjacent to the transport route (off-link exposures), (2) exposures to persons (e.g., passengers on passing trains or vehicles) sharing the transport route (on-link doses), (3) exposures to persons at stops (e.g., residents or rail and truck crew not directly involved with the shipment), and (4) exposures to transportation crew members (occupational). The exposures calculated for the first three groups were added together to estimate the general population exposure estimates for rail and truck transport; the exposure calculated for the fourth group represents occupational exposure to the rail crew exposures during inspections and truck crew during transit and inspections. Table A-6 summarizes the calculational methods used for each group for the shipment of naval spent nuclear fuel and test specimens.

As shown in Table A-6, simple calculations were performed to account for situations where the RADTRAN 4 computer code was not the best calculational model with respect to the transportation of naval spent nuclear fuel. The information used in the simple calculations was based on historical information. The results obtained using these simple calculations are expected to be equal to or greater than any exposures which might actually occur.

The maximum possible radiological exposure to an individual for the routine transport of naval spent nuclear fuel and test specimens off-site was estimated for transportation workers, as well as members of the general population. For rail shipments, the three general population scenarios were: (1) a railyard worker who might be working at a distance of 10 meters (32.8 feet) from the shipping container for 2 hours, (2) a resident who might live 30 meters (98.4 feet) from the rail line

Table A-6. Calculational methods used to obtain exposures for population groups of interest.

Shipment Type	Origin	Destination[a]	Mode	General Population			Occupational	
				Off-Link and On-Link	Stops	Maximum Individual	Workers	Escorts
Spent Nuclear Fuel to ECF or Equivalent	Kesselring Site	Ballston Spa	Truck	(1)	(3)	(6)	(3)	(3)
	Shipyard/Rail Siding	Various	Rail	(1)	(1)	(6)	(2)	(5)
	Windsor Site	Griffen Siding	Truck	(1)	(3)	(6)	(3)	(3)
	Pearl Harbor	Puget Sound	Ship	N/A	N/A	N/A	(4)	(4)
Spent Nuclear Fuel to Storage	ECF or Equivalent	Various	Truck	(1)	(1)	(6)	(1)	(1)
Test Specimen Assemblies	TRA	Various	Truck	(1)	(1)	(6)	(1)	(1)
Test Specimens	ECF or Equivalent	Bettis/ KAPL, etc.	Truck	(1)	(1)	(6)	(1)	N/A

Calculational Methods:

(1) RADTRAN 4 calculations.
(2) RADTRAN 4 rail calculations for inspection exposure and simple calculations based on rail transportation data supplied by the government escorts for rail transit exposure.
(3) Simple calculation model based on truck transportation data supplied by site personnel.
(4) Simple calculation model based on ship transportation data supplied by Pearl Harbor Naval Shipyard.
(5) Exposures based on historical TLD readings.
(6) Simple calculation model based on scenarios provided in RISKIND.
(a) The methods provided in this table apply to the destination for all the alternatives evaluated.

where the shipping container was being transported, and (3) a resident who could be living 200 meters (656.2 feet) from a rail stop where the shipping container was sitting for 20 hours. The government escorts and crew members from the rail, heavy-lift transporter, and ship were evaluated for the transportation workers (occupational). Based on records of past escorted rail shipments, the government escort might be the same individual for as many as two-thirds of the shipments in a 5-year period. The crew members were postulated to be the same individuals for all shipments in the 40-year period.

For off-site truck shipments, the three scenarios for the general population were: (1) a person who might be caught in traffic and located 1 meter (3 feet) away from the surface of the shipping container for one-half hour, (2) a resident who might be living 30 meters (98.4 feet) from the highway used to transport the shipping container, and (3) a service station worker who might be working at a distance of 20 meters (65.6 feet) from the shipping container for 2 hours. The hypothetical maximum exposed individual radiological exposures were accumulated over the 40-year period. However, for the situation involving an individual who might be caught in traffic next to a truck transporting spent nuclear fuel, the radiological exposures were only calculated for one event since it was considered unlikely that the same individual would be caught in traffic next to all containers for all shipments. For truck shipments, the occupational maximum exposed individual is the driver. For each of the categories of truck shipments described in Sections A.4.2 through A.4.4, the calculations used a single individual as the driver for all shipments made in the past. For shipments in the 40-year period being evaluated, a single person was also used in the calculations as the driver for all shipments of each category.

The hypothetical maximum exposed individual scenarios for the general population described above were not applicable for on-site shipments of naval spent nuclear fuel and test specimens for two reasons. The first is that there are no members of the general population in the vicinity during the on-site shipments. The second reason is that an obstruction, if encountered, would be safely avoided under the direction of the escorts. Two alternate scenarios were developed. They were: (1) a site employee in a disabled vehicle along the transport route, located 10 meters (32.8 feet) from the container and (2) a site employee trailing the slow-moving transport vehicle for the entire trip. These scenarios were considered to be single-event occurrences.

As noted in Table A-6, simple methods were also used to calculate radiological exposures. For radiological exposures to personnel at a fixed distance from the shipping container, the following equation was used.

<u>Exposures to personnel at a fixed distance from the container:</u>

$$= N \times NBA \times T \times SF \times K \times TI \;/\; D^2$$

where:

N	=	number of people
NBA	=	factor to account for exposure decrease at increased distance from the source (attenuation/buildup). (Refer to Neuhauser and Kanipe 1993.)
T	=	time
SF	=	shielding factor
K	=	transport index to exposure rate conversion factor
TI	=	transport index (see Section A.7.1.1.2)
D	=	distance from the centerline.

For the radiological exposures associated with the ship transport of spent nuclear fuel from the Pearl Harbor Naval Shipyard to the Puget Sound Naval Shipyard, the following general equations were used:

<u>Exposures to personnel aboard ship during transport:</u>

$$= N \times NBA \times T \times SF \times K \times TI \times (1/(X_1 + X_2)^2 + 1/X_2^2)$$

where:

X_1	=	distance between the centerlines of the two shipping containers
X_2	=	distance between centerline of the nearest shipping container and the exposed individual

<u>Exposures to personnel aboard ship during inspections:</u>

$$= (N \times T \times TI) + (N \times NBA \times T \times K \times SF \times TI \;/\; (X_1 - R - 1)^2)$$

where:

R = effective radius to account for the exposure from the second shipping container.

Table A-7 provides an estimate of the number of people included in the analyses. To determine this number, the basic equation used was:

(Distance Traveled) x (Exposure Path Width) x (Density of People).

In each alternative, there are many shipments from several different origin/destination combinations. Since the route would be the same for each shipment from the same origin/destination combination, the people along the route would also not change, therefore, the distance used was from one trip for each origin/destination combination. The exposure path width is 1.6 kilometers (1 mile), consistent with the RADTRAN 4 computer code methodology for incident-free calculations. The population density was calculated by summing the product of the fraction of travel times the density in each population area (rural, suburban, and urban). The fraction of travel and density were obtained from HIGHWAY and INTERLINE. The total number of people was then calculated by summing the results of all origin/destination combinations for each alternative.

Table A-7. Estimated number of people included in incident-free transportation analyses.

Alternative	Number of People
No Action	890,000
Decentralization - No Examination	890,000
Decentralization - Limited Examination	9,240,000
Decentralization - Full Examination	6,820,000
1992/1993 Planning Basis	7,290,000
Regionalization or Centralization at INEL	7,290,000
Regionalization or Centralization at Hanford	8,370,000
Regionalization or Centralization at Savannah River	6,950,000
Regionalization or Centralization at Oak Ridge	5,660,000
Regionalization or Centralization at Nevada Test Site	8,320,000

A.5.2 Technical Approach for Transportation Accidents

The RADTRAN 4 computer code was used to calculate the radiological risk to the general population and transportation (occupational) crew under accident conditions. The RADTRAN 4 computer code evaluates six pathways for radiation exposures resulting from an accident. The six potential pathways are:

- Direct Radiation Exposure from the Damaged Container
- Inhalation Exposure from the Plume of Radioactive Material Released from the Damaged Container
- Direct Radiation Exposure from Immersion in the Plume of Radioactive Material Released from the Damaged Container
- Direct Radiation Exposure from Ground Deposition of the Radioactive Material Released from the Damaged Container
- Inhalation Exposure from Resuspension of the Radioactive Material Deposited on the Ground
- Ingestion Exposure from Food Products Grown on the Soil Contaminated by Ground Deposition of Radioactive Material Released from the Damaged Container.

For each pathway, a specific formula is used to determine an estimate of the radiological risk, expressed in exposure, from that particular pathway with the total radiation exposure equal to the sum of the exposure for each pathway. The total accident radiation exposure accounts for the probability of an accident occurring and the probability of an accident of a particular severity. It should be noted that all consequences are included in the risk assessment, regardless of the probability. The general equation for the population exposure from all pathways is:

$$D_R = \sum_{c,r} (N_c \times L_{r,c} \times P_r \times \sum_{i,j,k} (P_j \times RF_j \times D_{i,j,k}))$$

where: D_R = population exposure from the accident

N_c = number of naval spent nuclear fuel modules shipped of fuel type c

$L_{r,c}$ = shipment distance for fuel type c shipped through state r

P_r = frequency of traffic accidents

P_j = probability of occurrence of accident severity category j

RF_j = fraction of curies released from shipping container by severity category j

$D_{i,j,k}$ = radiation exposure resulting from accident severity category j through pathway i in population density zone k.

The accident risk evaluation was performed using neutral and stable atmospheric conditions (Pasquill Stability Classes D and F, respectively). The neutral atmospheric condition results provide a best estimate of the risk. Stable atmospheric conditions resulted in values approximately twice the neutral conditions, ignoring the lower probability of occurrence.

In addition to the estimation of the radiological risk of an accident described above, an evaluation of the consequences of an accident of the highest severity was performed. The consequences, expressed as radiological exposure, are calculated for the maximum exposed individual and the general population. Exposures to the general population were calculated for each of the three population density regions (rural, suburban, and urban). The maximum exposed individual was placed in the population area which resulted in the highest exposure.

The RISKIND computer code, modified by its authors to accept the fission product inventory unique to naval spent nuclear fuel, was used to calculate the maximum consequences. The pathways evaluated by RISKIND are identical to those used in the RADTRAN 4 computer code for the risk evaluation.

The maximum consequence evaluation presents the consequences for design basis accidents, defined as those accidents which have a probability of greater than 1×10^{-6} per year, and beyond design basis accidents, defined as those which have a probability of 1×10^{-6} to 1×10^{-7} per year. Accidents with a probability of less than 1×10^{-7} were not analyzed in the maximum consequence evaluation.

To determine the overall probabilities, the probability of an accident, the probability of the consequences, fraction of travel in each population area, and probability of the meteorological conditions had to be determined.

The probability of the accident was calculated by multiplying the accident rates for each state times the distance traveled in each state times the number of shipments. The results were summed for each combination of origin and destination for the alternative.

As described later in Section A.7, a study performed by Lawrence Livermore National Laboratory entitled "Shipping Container Response to Severe Highway and Railway Accident Conditions" (NUREG 1987) grouped accidents into categories by strain and container mid-wall temperatures and calculated the probabilities of accidents of each category. Section A.7 also describes the consequences associated with each accident category for the naval spent nuclear fuel and test specimen shipments. The probabilities were summed for the categories which have the same consequences.

The fraction of travel in each population area (rural, suburban, and urban) was obtained from INTERLINE and HIGHWAY for each origin/destination combination. Each alternative consists of many shipments from various origin/destination combinations; therefore, an overall fraction was calculated. The overall fraction, by alternative, was calculated by multiplying each origin/destination fraction (from INTERLINE and HIGHWAY) by the number of shipments from that particular origin/destination combination, summing the results and dividing by the total number of shipments.

To calculate the probability of the meteorological conditions, Pasquill Class D was considered to be equivalent to 50% meteorology; that is, 50% of the time, conditions are expected to be more severe, and 50% of the time, conditions are expected to be less severe. Pasquill Class F was considered to be equivalent to 95% meteorology; that is, 5% of the time, it is more severe, and 95% of the time, it is less severe. Since the difference in 50% (1 chance in 2) and 95% (1 chance in 20) is a factor of 10, the probability of encountering Pasquill Class F was concluded to be a factor of 10

less than Pasquill Class D. Analyses performed by the National Oceanic and Atmospheric Administration (Doty et al. 1976) confirm that this assumption is reasonable.

The overall probability of the consequence of an accident for each population area was then calculated by multiplying the accident probability times the consequence probability times the fraction of distance traveled. Starting with the highest consequences, the probabilities were then compared to the 1×10^{-6} per year criterion for the design basis accidents and 1×10^{-7} per year criterion for the beyond design basis accidents. If the probability was greater than 10 times the criterion (1×10^{-6} or 1×10^{-7}), the most severe Pasquill Class F results were presented. If not, and the probability was greater than the criterion (1×10^{-6} or 1×10^{-7}), Pasquill Class D was presented. If the probability was less than the cutoff, the probabilities having the next most severe consequences were compared to the same criterion and this step was repeated until all consequences were evaluated. As a minimum, the consequences resulting from release of 1% of the corrosion products (Pasquill Class D) were presented.

Careful attention was paid to ensure that the probabilities were not calculated for such small categories that the resulting probabilities were less than the criterion and results would inadvertently present less severe consequences. When the highest consequence accident did not meet the criterion, the probability of the next highest accident was determined by summing both the accident consequence being evaluated and the probability of the higher consequence accidents previously shown to have a probability less than the criterion. This same technique was applied to the fraction of travel (urban fraction is equivalent to highest consequence, suburban fraction is next highest, etc.) as demonstrated in the following example.

Probability of the accident of Consequence A	-	1.17×10^{-7}
Fraction of distance traveled in rural area	-	0.85
Fraction of distance traveled in suburban area	-	0.11
Fraction of distance traveled in urban area	-	0.04

The urban fraction was multiplied by the probability, and the resultant probability of an accident of Consequence A in an urban area was 4.68×10^{-9}. The consequences of this accident would not be evaluated. For the suburban area, the suburban and urban fractions were added and then multiplied by the probability (1.75×10^{-8}). Again, the consequences of this accident would not be evaluated since the probability is less than 1×10^{-7}. Likewise, for the rural area, the rural, suburban, and urban fractions were added and multiplied by the probability. Using this technique, the probabilities would indicate that the rural probability was 1.17×10^{-7}, which is greater than the 1×10^{-7} criterion and the Consequence A results would be presented. If the fractions were used at face value, however, the probability of an accident of Consequence A would have been 4.68×10^{-9} in an urban area, 1.29×10^{-8} in a suburban area, and 9.95×10^{-8} in a rural area. When individually compared to the 1×10^{-7} criterion, this accident would not have been presented for any area.

Accident results are presented for both the maximum exposed individual and the general population. These results include members of the transportation crew.

A.6 ROUTING ANALYSIS

In order to assess the radiological risks associated with transportation, it was necessary to determine route characteristics based on the origin and destination of each shipment.

For naval spent nuclear fuel shipments, the origin is the prototype or shipyard location where the naval spent nuclear fuel is removed from a prototype or shipboard reactor. The destination is ECF, Savannah River Site, Hanford Site, Oak Ridge Reservation, Nevada Test Site, or Puget Sound Naval Shipyard, depending on the alternative. For each origin and destination pair, the potential rail routes have been generated and analyzed using the INTERLINE computer code (Johnson 1993a). For shipments originating from Pearl Harbor Naval Shipyard, the containers travel by ship to Puget Sound Naval Shipyard, where they are transferred to rail for shipment to the destination following the same routes as the naval spent nuclear fuel shipments originating from Puget Sound Naval Shipyard. The shipment travel time by ocean was based on historical data on the time in transit, independent of the actual route. For heavy-lift transporter shipments from the Kesselring and Windsor prototype sites to the closest rail siding, the actual street routes and shipment duration times based on previous shipments were used.

INTERLINE is an interactive computer program designed to simulate routing using the U.S. rail system. The INTERLINE code used is the latest available from Oak Ridge National Laboratory and contains the 1990 census data. The INTERLINE data base consists of networks representing various competing rail companies in the U.S. The routes used for the transportation evaluation use the standard INTERLINE model which simulates the selection procedure that railroad companies would use to direct shipments of spent nuclear fuel. The code is updated periodically to reflect

current track conditions and has been benchmarked against reported mileages and observations. INTERLINE also provides the weighted population densities for rural, suburban, and urban populations for each state and averaged over all states along the shipment route and the percentage of mileage traveled in each population density. The distance traveled, weighted population density, and percentage of distance in each population density are input variables in the RADTRAN 4 code.

For the off-site transportation of the test specimen assemblies and test specimens, all shipments are made by exclusive-use truck which includes no other freight. The destinations are ECF, Savannah River Site, Hanford Site, Oak Ridge Reservation, Nevada Test Site, Puget Sound Naval Shipyard, Bettis Atomic Power Laboratory, and Knolls Atomic Power Laboratory for the various alternatives. For each origin and destination pair, the potential truck routes have been generated and analyzed using the routing model HIGHWAY.

HIGHWAY is an interactive computer code designed to simulate routing using the U.S. highway system. The HIGHWAY code used for this report is the latest available from Oak Ridge National Laboratory. The code is updated periodically as new roads are added. HIGHWAY provides the distance between the origin and destination, the weighted population densities along the route, and the percentage of distance traveled in each population density, all input variables for the RADTRAN 4 computer code.

For the on-site transportation, HIGHWAY only has two of the sites on the INEL. This origin/destination pair was run using HIGHWAY to determine the population densities and percentage of travel in each population density. The actual distance between sites on the INEL was measured.

A.7 INPUT PARAMETERS

The major input parameters and models used to evaluate the radiological risks associated with the five alternatives described in Section A.3 are provided in this section. Standard RADTRAN 4 computer code values, as well as actual data gathered from historical naval spent nuclear fuel and test specimen shipments, were used as the basis for the input parameters. For those situations where historical data were available, the actual data were used in place of the standard RADTRAN 4 computer code values to provide the best estimate of the radiological risks associated with each alternative.

A.7.1 Shipments of Naval Spent Nuclear Fuel from Shipyards and Prototypes

A.7.1.1 Incident-free Transportation of Spent Nuclear Fuel from Shipyards and Prototypes. This section provides the input parameters used to determine the radiological impacts associated with the routine, incident-free (i.e., no accident) transportation of spent nuclear fuel for each of the five alternatives.

A.7.1.1.1 Planned Shipments. The list of planned shipments of naval spent nuclear fuel by origin is provided in Table A-8.

Table A-8. Planned shipments of naval spent nuclear fuel from shipyards and prototypes.

	Generating Site				Origin or
Alternative	East Coast	West Coast	NRF	TOTAL	Destination
No Action, Decentralization - No Exam	204	0	0	204	To Norfolk
Decentralization - Limited Exam	53 181 234	0 0 0	1 0 1	54 181 235	To Puget Sound To Norfolk
Decentralization - Full Exam	314 314 628	261 261 522	0 0 0	575 575 1150	To ECF From ECF
1992/1993 Planning Basis, Regionalization at INEL and Centralization at INEL	314	261	0	575	To ECF
All other Regionalization and Centralization Alternatives	314	261	3	578	To Regionalization or Centralization site

A.7.1.1.2 Transport Index. Historical information from prior shipments was used to estimate the expected external radiation exposure rates for future shipments. This information included actual measured radiation levels and the recorded Transport Indexes (TIs) from past shipments. The TI used in this analysis is the sum of the maximum neutron and gamma radiation measured at 1 meter (3.3 feet) from the surface of the cask. The TIs that were used ranged from 0.1 to 1.8.

A.7.1.1.3 Transportation Distances and Population Densities. Section A.6 provided a description of the general methodology used for determining transportation distances and the population densities along the transportation routes. Historical data were obtained on the distance traveled for shipments from the shipyards and prototype sites to ECF. These data were averaged by origin and compared to the value calculated by INTERLINE. The actual data were approximately 11% higher than the distance predicted by INTERLINE on average. In order to provide the best estimate exposure, which is based on the distance traveled, the INTERLINE distances were increased by 11% for the 1992/1993 Planning Basis alternative. One of the primary reasons the actual distances traveled were judged to be longer than the INTERLINE prediction was the escort responsibility to avoid potential delays due to track or security problems. The shipments to the alternative sites will also be escorted and therefore the same increased travel distance is expected. The 11% increase in distance traveled was also applied to all other alternatives. This technique allowed for comparison of the alternatives on an equal basis. The percentages of distance traveled in each population density calculated by INTERLINE were applied to the distances increased by 11%.

A.7.1.1.4 Train Speed. The RADTRAN 4 computer code provides standard values for train speeds that are dependent on the population density. For rural areas, the standard value is 64.4 kilometers per hour (40 miles per hour (mph)). For suburban areas, the standard value is 40.2 kilometers per hour (25 mph), and for urban areas, the standard value is 24.1 kilometers per hour (15 mph). However, naval spent nuclear fuel shipments are required to be transported at speeds not to exceed 56.3 kilometers per hour (35 mph). Government escort logs from historical spent nuclear fuel shipments support use of 24.1 kilometers per hour (15 mph). This 24.1 kilometers per hour (15 mph) train speed estimate was used to evaluate all five alternatives.

A.7.1.1.5 Train Stop Time. The RADTRAN 4 computer code provides standard values for train stop times that are either dependent or independent of the distances traveled. For naval spent nuclear fuel transported by rail, the government escorts are responsible for ensuring that the shipments are made in the most efficient and safe manner. The government escort logs for historical spent nuclear fuel shipments were reviewed, and actual stop times were determined to be much shorter than the standard RADTRAN 4 computer code values. The recorded stop times were divided by the actual distance traveled from historical data over the last 3 years and an average of 0.02 hour per kilometer (0.032 hour per mile) was calculated. This value was used to evaluate all five alternatives since the rail transportation of spent nuclear fuel will always be accompanied by government escorts and all alternatives originate from the same locations.

A.7.1.1.6 Number of Train Crew Members. The standard RADTRAN 4 computer code value for the number of train crew members is five. For all shipments to NRF, all rail companies with the exception of Burlington Northern have two crew members during shipments, located in the locomotive. Burlington Northern adds a third crew member in a caboose immediately behind the government escort caboose. In the RADTRAN 4 computer code, exposure to the crew members is not calculated since the distance to the crew members is large. In actuality, the distance to the Burlington Northern crew member located in the caboose is less than that used in the RADTRAN 4 computer code and therefore simple calculations were performed to determine the radiological exposure. In addition, naval spent nuclear fuel shipments also are shipped periodically by "special train." In the special train configuration, the two crew members in the locomotive are one car from the railcar with the shipping container. Historically, these shipments occur approximately 42 percent of the time. The majority of shipments by "special" train are arranged by the railroad companies to meet railroad schedules. On occasion, the Navy requests "special" train service for shipments with high-priority examination material. Simple calculations were also performed to determine the radiological exposure during these special shipments. For shipments to the sites other than NRF, there was no experience with all railroad companies which would have to be used; however, there is no reason to expect the rail companies to change their standard practices. In these cases, there would be two train crewmen, both located in the engine area. Forty-two percent of the shipments would be shipped by special train to the alternate sites. When applicable, the third Burlington Northern crew member was also accounted for.

A.7.1.1.7 Transport Index to Exposure Rate Conversion Factors. Container transport index to exposure rate conversion factors for the M-130 and M-140 shipping containers were calculated using the standard equation in the RADTRAN 4 computer code. The results were compared to detailed computer analyses performed using SPAN4, and the RADTRAN 4 results were found to overestimate the exposure by a factor of two to three. Using the SPAN4 computer code results, the effective package dimensions of the containers used in the RADTRAN 4 calculations were adjusted to provide a conservative yet more realistic value of the transport index to exposure rate conversion factor. Due to similarities in the construction and fuel shipped, the M-130 conversion factor was applicable to the M-160. The values used are provided in Table A-9.

Table A-9. Transport index to exposure rate conversion factors for the M-130, M-140, and M-160 shipping containers.

Container	Effective Package Dimension (meters)	Transport Index to Exposure Rate Conversion Factor
M-130/M-160	2.50 (8.2 feet)	5.06
M-140	3.20 (10.5 feet)	6.76

A.7.1.1.8 Train Stop Shield Factors. For train stops, the standard RADTRAN 4 computer code gamma and neutron radiation shield factors are both assigned as 0.1. This value includes the presence of substantial railyard steel structures equivalent to approximately 4 inches of steel. Four inches of steel reduces gamma radiation by more than a factor of 10; however, the steel only reduces neutron radiation by a factor of approximately 2. Therefore, a shield factor of 0.5 was conservatively used for neutron radiation. In order to incorporate this shielding into the RADTRAN 4 computer code, separate gamma and neutron radiation exposure calculations were performed. However, since RADTRAN 4 does not permit separate shielding factors to be used for different types of radiation, the stop times for the neutron radiation evaluations were increased by a factor of 5 to provide an equivalent increase in neutron exposure. These more realistic changes to the standard RADTRAN 4 computer code values were incorporated for all five alternatives.

A.7.1.1.9 Radiation Exposure Decrease Due to Distance. The RADTRAN 4 computer code provides standard values for determining the gamma and neutron radiation exposure decrease at increasing distance from the source. For gamma radiation, the RADTRAN 4 computer code uses the $1/x^2$ decrease due to distance. The RADTRAN 4 computer code also specifically calculates the decrease in neutron exposure at increased distances. The adequacy of the RADTRAN 4 radiation exposure decrease was evaluated. The gamma radiation decrease factor used by RADTRAN 4 was consistent with the results predicted for naval fuel. The RADTRAN 4 prediction for neutron radiation slightly overpredicts the decrease in exposure at far distances for the shipping containers used for naval shipments. Using the same basic equation used by RADTRAN, a value of 2.0×10^{-10} was used for the RADTRAN 4 constant a_4 in lieu of 0. The value of 2×10^{-10} produces results which are slightly higher than the standard method and agree with measurements of neutron exposure rates from naval spent nuclear fuel shipments.

A.7.1.1.10 Shipment Storage Time. As noted previously, the government escorts accompanying the rail shipments of spent nuclear fuel are responsible for ensuring that the naval spent nuclear fuel shipments are made in the most efficient and safe manner. Naval spent nuclear fuel is not stored while being shipped; therefore, there was no intermediate shipment storage time associated with any of the alternatives. There is also no intermediate storage time during the heavy-lift transport shipments from the prototype sites and the ocean shipments from Pearl Harbor Naval Shipyard.

A.7.1.1.11 Heavy-lift Transporter Transportation Crew. Information from records of naval spent nuclear fuel shipments was reviewed to determine a realistic estimate of the number of people involved, the amount of time required, and the distances between individuals and the shipping container. The number of hours worked ranged from 1 to 10 and the distance from the container ranged from 1.5 to 91 meters (5 to 300 feet). For simplicity, weighted averages of the number of hours and distances from the shipping container were calculated and are provided in Table A-10.

Table A-10. Summary of the number of people involved and distance from the container during heavy-lift transporter shipments to the rail siding at the prototype sites.

Prototype	Number of People	Number of Hours per Worker	Distance from the Shipping Container (meters)
Windsor Site	37	5.08	25.0 (82 feet)
Kesselring Site	36	5.11	32.3 (106 feet)

This information was used to evaluate all five alternatives.

A.7.1.1.12 Time to Ship by Heavy-lift Transporter. Based on discussions with personnel at the prototype facilities who have made shipments and a review of records, the average duration of the heavy-lift transporter shipment from the prototype sites to the local rail siding is 2 hours.

A.7.1.1.13 Number of Heavy-lift Transporter Inspections. The shipments are inspected prior to leaving the prototype's site boundaries, and no additional inspections are performed during the short heavy-lift transporter shipment. As a result, there are no inspections during the heavy-lift transporter shipment in the evaluation of the five alternatives.

A.7.1.1.14 Heavy-lift Transporter Stop Time. Shipments of spent nuclear fuel from the two prototype locations are first transported by heavy-lift transporter to the nearest rail siding. Information from records of naval spent nuclear fuel shipments was reviewed to determine a realistic estimate of the heavy-lift transporter stop times. For naval spent nuclear fuel heavy-lift transporter shipment from the Windsor Site, a heavy-lift transporter stop time of 24 hours was used. For heavy-lift transporter shipments from the Kenneth A. Kesselring Site, a stop time of 10 hours was used. The heavy-lift transporter shipments from the prototypes to the rail sidings occur through suburban populations only. These heavy-lift transporter stop times were used to evaluate all five alternatives.

A.7.1.1.15 Standard RADTRAN 4 Computer Code Values Used. The following standard RADTRAN 4 computer code value was reviewed and determined to reflect the best estimate of current railroad industry practice:

- Number of Inspections of the Shipping Container and Railcar.

The following standard RADTRAN 4 computer code estimates of the populations that could be affected by the shipment of spent nuclear fuel were also used for the five alternatives:

- Number of People per Vehicle Sharing the Transport Route (On Link)
- Traffic Count Passing a Specific Point - Rural, Suburban, and Urban Zones
- Average Exposure Distance When Stopped
- Persons Exposed While Stopped
- Fraction of Travel During Rush Hour, on City Streets, and on Freeways.

A.7.1.1.16 Number of Ship Inspections. Shipments of spent nuclear fuel from Pearl Harbor Naval Shipyard must first be transported by ship to the Puget Sound Naval Shipyard. Using the standard values in the RADTRAN 4 computer code, the radiological exposures to the crew and government escorts are negligible since the distances from these individuals to the shipping containers are large. As a result, the radiological exposure estimates are only expected to occur during inspections. Based on radiation monitoring results for past naval spent nuclear fuel shipments, this is not realistic for naval spent nuclear fuel, and a separate calculational model was developed to account for this potential radiation exposure. The model uses the standard point source formula (see Section A.5.1) to calculate the crew and government escort exposures during transport by ship. The model took into account the ship used, transport index, transport time, distance between shipping containers, distance from the shipping containers and living quarters, distance from the shipping containers and the engine room, the number of crew members and government escorts, and the time required for inspections based on records from historical shipments of spent nuclear fuel. After reviewing historical shipment records, it was determined that three different sized ships have recently been used. The smallest one, Ship 1, was used once and is not expected to be used in the future. Only the other two, Ships 2 and 3, would be used in the future, in equal proportion. Table A-11 below provides the information used to calculate the radiological exposures resulting from transporting naval spent nuclear fuel by ship. This model was used to evaluate all five alternatives.

Table A-11. Parameters used to calculate crew and escort exposure during ocean travel from Pearl Harbor Naval Shipyard to Puget Sound Naval Shipyard.

Parameter	Ship 1	Ship 2	Ship 3
Transport Time, T, in days	11	8	9
Separation Between M-130s, X_1, in feet	92	43	20
Nearest Distance to Living Quarters, X_2, in feet	40	80	300
Nearest Distance to Engine Room, X_3, in feet	20	80	300
Number of Crew Members, N_c	11	22	26
Number of Government Escorts (not part of crew size), N_e	2	2	2
Escort Inspection Time (per Escort), in hr/day	0.50 for historic 0.25 for future		
Shielding Factor	(1/3) for gamma, (2/3) for neutron, for every 40-foot increment from the container centerline		

A.7.1.2 Accident During Transportation of Spent Nuclear Fuel. This section provides the input parameters used to calculate the radiological impacts for accidents during transportation of spent nuclear fuel for evaluation of the five alternatives. The planned shipments, transportation distances, population densities, and the percentages of travel in each population density described in Section A.7.1.1 were also used for the accident analyses. Unless otherwise described in this section, the standard values provided by the RADTRAN 4 and RISKIND computer codes were used.

A.7.1.2.1 Accident Probability. The probability of a rail accident used for evaluation of all alternatives was obtained from "Longitudinal Review of State-Level Accident Statistics for Carriers of Interstate Freight" (Saricks and Kvitek 1994). The probabilities are provided both by state and a national average. The state dependent probabilities were used for the accident risk assessment. Past naval spent nuclear fuel shipments have traveled approximately 2 million kilometers (1.24 million miles) by rail without an accident, which is consistent with the national average of 5.57×10^{-8} accident per kilometer.

A.7.1.2.2 Accident Severity Categories and Probabilities. In the "Shipping Container Response to Severe Highway and Railway Accident Conditions" (NUREG 1987), referred to as the "Modal Study," Lawrence Livermore National Laboratory categorized the potential damage to shipping containers according to the magnitude of the thermal and mechanical forces that could result from an accident. The structural and thermal forces were categorized into 20 regions. Given that an accident occurs, the probability that the accident would be in each region was calculated for both rail and truck shipments. Table A-12 provides the probabilities for rail accidents by region.

Table A-12. Accident severity probabilities for rail shipments.

Structural Response (maximum strain on inner shell, %)					
	R(4,1) 1.786×10^{-9}	R(4,2) 3.290×10^{-13}	R(4,3) 2.137×10^{-13}	R(4,4) 1.644×10^{-13}	R(4,5) 3.459×10^{-14}
S_3 (30)					
	R(3,1) 5.545×10^{-4}	R(3,2) 1.0217×10^{-7}	R(3,3) 0.634×10^{-8}	R(3,4) 5.162×10^{-8}	R(3,5) 5.296×10^{-8}
S_2 (2)					
	R(2,1) 2.7204×10^{-3}	R(2,2) 5.011×10^{-7}	R(2,3) 3.255×10^{-7}	R(2,4) 2.531×10^{-7}	R(2,5) 1.075×10^{-8}
S_1 (0.2)					
	R(1,1) 0.993962	R(1,2) 1.2275×10^{-3}	R(1,3) 7.9511×10^{-4}	R(1,4) 6.140×10^{-4}	R(1,5) 1.249×10^{-4}
	T_1 (500)	T_2 (600)	T_3 (650)	T_4 (1050)	

Thermal Response (lead mid-thickness temperature, °F)

A.7.1.2.3 Naval Spent Nuclear Fuel Integrity Following an Accident. Detailed structural and thermal analyses were performed for the shipping containers used for naval spent nuclear fuel shipments up to an equivalent strain of 30% and mid-wall temperature of 1050°F. For these cases, the naval spent nuclear fuel was not damaged. For the thermal and structural regions above 1050°F and 30% strain, the modal study defines the upper limits as unbounded. The naval spent nuclear fuel was postulated to be damaged and the fission products and corrosion products would be released in the quantities described in Table A-13 for the risk analyses.

A.7.1.2.4 Release Fractions. The release fractions were derived based on the results presented in the NRC modal study (NUREG 1987) and the results of the structural and thermal analyses described above. Although the naval spent nuclear fuel is stronger, the release fractions for the boiling water reactor (BWR), pressurized water reactor (PWR), and aluminum-clad fuel from the modal study were used. From the modal study, the release fraction in lower left region R(1,1) is zero for the risk evaluation. For the maximum consequence evaluation, 1% of the corrosion products might be released for the lower left region, R(1,1). Based on the results of the structural and thermal analyses up to 30% strain and 1050°F mid-wall temperature, the naval spent nuclear fuel is not damaged; therefore, regions R(1,2), R(1,3), R(2,1), R(2,2), R(2,3), R(1,4), R(2,4), R(3,4), R(3,1), R(3,2) and R(3,3) do not release fission products. Ten percent of the corrosion products might be released. In the remaining regions, 10% of the fission products might be available for release and released at the fractions specified below, also using a release of 10% of the corrosion products. Table A-13 provides the release fractions used.

Table A-13. Cask release fractions used for the RADTRAN 4 risk analyses.

	Release Fraction[a]					
Cask Response Region	Inert Gas	Iodine	Cesium	Ruthenium	Particulates	Corrosion Products
R(1,1)	0.0	0.0	0.0	0.0	0.0	0.0
R(1,2), R(1,3)	0.0	0.0	0.0	0.0	0.0	1.0
R(2,1), R(2,2), R(2,3)	0.0	0.0	0.0	0.0	0.0	1.0
R(1,4), R(2,4), R(3,4)	0.0	0.0	0.0	0.0	0.0	1.0
R(3,1), R(3,2), R(3,3)	0.0	0.0	0.0	0.0	0.0	1.0
R(1,5), R(2,5), R(3,5) R(4,5), R(4,1), R(4,2) R(4,3), R(4,4)	6.3×10^{-1}	4.3×10^{-2}	2.0×10^{-3}	4.8×10^{-4}	2.0×10^{-5}	1.0

[a] The release fraction represents the fraction of the fuel inventory available for release in the shipping container that would be released into the atmosphere following an accident of the given severity.

A.7.1.2.5 Plume Release Height. For the accident risk assessment, a ground level release was used. For the maximum consequence assessment, a plume release height of 10 meters (32.8 feet) was used.

A.7.1.2.6 Direct Exposure from a Damaged Shipping Container. A radiation level following the accident at the 10CFR71 regulatory limit of 1 rem at 1 meter (3.3 feet) from the container surface was used.

A.7.1.2.7 Food Transfer Factors. Food transfer factors were derived for the isotopes related to naval spent nuclear fuel in accordance with the methods described in Nuclear Regulatory Commission Guide 1.109 (NUREG 1977).

A.7.1.2.8 Distance from the Accident Scene to the Maximum Exposed Individual. No shielding was accounted for as the plume passes for the calculation of the exposure to the maximum individual. This location was determined using RISKIND based on the atmospheric stability and plume release height used. The maximum exposed individual could be a member of the rail crew or the general population.

A.7.1.2.9 RISKIND Population Density. The standard national average for each population density from the RADTRAN 4 computer code was used for the RISKIND maximum consequences assessment (6 people per square kilometer for rural, 719 for suburban, and 3861 for urban).

A.7.1.2.10 Radionuclide Inventory. The amount of radionuclides which would be released from an average shipment are provided in Table A-14. The values factor in the damage fraction described in Section A.7.1.2.3 and release fractions described in Section A.7.1.2.4. The radionuclides listed result in 99 percent of the exposure in all pathways.

Table A-14. Radionuclides which would be released from an average shipment of naval spent nuclear fuel from a shipyard or prototype.

For Accidents which Release Both Fission and Corrosion Products		For Accidents which Release Only Corrosion Products	
Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Kr-85	9.85×10^{2}	Co-58	1.61×10^{-1}
Cs-134	3.72×10^{1}	Mn-54	2.22×10^{-2}
Cs-137	3.44×10^{1}	Fe-55	6.62×10^{-1}
H-3	1.39×10^{1}	Co-60	3.63×10^{-1}
Ru-106	9.02×10^{-1}	Sr-90	3.14×10^{-4}
Ce-144	4.89×10^{-1}	Ni-63	1.19×10^{-1}
Co-60	3.63×10^{-1}		
Sr-90	3.41×10^{-1}		
Pu-238	1.02×10^{-2}		
Pu-241	3.43×10^{-3}		
Cm-244	1.36×10^{-4}		

A.7.2 Transfers of Naval Spent Nuclear Fuel to Storage Following Examination

A.7.2.1 Incident-free Transportation of Naval Spent Nuclear Fuel to Storage. This section provides the input parameters used to determine the radiological impacts associated with the routine, incident-free (i.e., no accident) transportation of naval spent nuclear fuel to storage for each of the five alternatives.

A.7.2.1.1 Planned Shipments. Table A-15 provides the number of planned transfers in each cask.

Table A-15. Planned transfers of naval spent nuclear fuel to storage.

	NFS-100	Peach Bottom	Large Cell
No Action, Decentralization - No Exam, Decentralization - Limited Exam	0	0	15
Decentralization - Full Exam	0	0	14
1992/1993 Planning Basis, All Regionalization Alternatives, All Centralization Alternatives	196	64	468

A.7.2.1.2 Transport Index (TI). A TI of 0.3 was used for all NFS-100 cask transfers. This value was determined from recorded measurements over the last 3 years for the same fuel types planned to be transferred in the future. The Peach Bottom and Large Cell casks have not previously been used for the planned transfers and therefore historic data were not available. Based on a comparison of predicted TI values from conservative safety analyses to the actual measured TI's for similar casks and fuel types, a TI of 1.0 was calculated for both the Peach Bottom and Large Cell casks.

A.7.2.1.3 Transportation Distances and Population Densities. Section A.6 provided a description of the general methodology used for determining transportation distances and the population densities along the transportation routes. The distance between ECF and ICPP is 9.7 kilometers (6 miles). From the HIGHWAY computer code, the transfer of naval spent nuclear fuel to storage occurs in a rural area. As stated in Section A.3.5, the storage facility at the alternative sites was identical to ICPP. Therefore, for the evaluation of the alternatives, the distance traveled and population density of the ECF to ICPP transfer were also used for the evaluation of the other alternatives.

A.7.2.1.4 Truck Speed. The standard RADTRAN 4 computer code speed for truck shipments in a rural population is 88.5 kilometers per hour (55 miles per hour). One of the reasons an on-site worst credible accident is less severe than the 10CFR71 hypothetical accident is that the speed is severely limited by the on-site transportation procedures. An average speed of 24.1 kilometers per hour (15 miles per hour) was used.

A.7.2.1.5 Truck Stop Time. The standard RADTRAN 4 computer code provides values for truck stop times that are either dependent or independent of the distances traveled. The logs for historical transfers of naval spent nuclear fuel to storage were reviewed, and it was determined that the actual stop times (10 minutes) were much shorter than the standard RADTRAN 4 computer code values. A stop time of 10 minutes was used to evaluate all five alternatives.

A.7.2.1.6 Radiation Exposure Decrease Due to Distance. The radiation exposure decrease due to distance described in Section A.7.1.1.9 was also applied to the truck transfers of naval spent nuclear fuel to storage.

A.7.2.1.7 Distance from Source to Crew. A distance of 6.1 meters (20 feet) was measured between the shipping cask and the driver for the exclusive-use truck transfers of naval spent nuclear fuel shipments to storage. Two escorts, one located approximately 46 meters (150 feet) in front and one the same distance behind the transport vehicle, are also present. These data were used in the RADTRAN analyses for all alternatives.

A.7.2.1.8 Transport Index to Exposure Rate Conversion Factors. Transport index to exposure rate conversion factors for the casks used for transfers of naval spent nuclear fuel to storage were calculated using the standard equation in RADTRAN 4. The results were compared to detailed computer analyses performed using SPAN4, and RADTRAN 4 results were found to overestimate the exposure. Using the SPAN4 computer code results, the effective package dimensions of the casks used in the RADTRAN 4 calculations were adjusted to provide a conservative yet more realistic value of the transport index to exposure rate conversion factor. The values used are provided in Table A-16.

Table A-16. Transport index to exposure rate conversion factors for the NFS-100, Peach Bottom, and Large Cell casks.

Cask	Effective Package Dimension (meters)	Transport Index to Exposure Rate Conversion Factor
NFS-100	3.8 (12.5 feet)	8.41
Peach Bottom	2.8 (9.2 feet)	5.76
Large Cell	3.2 (10.5 feet)	6.76

A.7.2.1.9 Storage. There is no intermediate storage time during transfers of naval spent nuclear fuel to its destination.

A.7.2.1.10 Persons Exposed While Stopped. The only stop time for the transfer of naval spent nuclear fuel to storage occurs during routine surveys at the destination entrance. This area is well removed from highway and general population and therefore no people were considered to be exposed during the short 10-minute stop. The escorts are not present during the surveys and the driver remains in the cab of the truck, 6.1 meters (20 feet) from the cask during the surveys. The people performing the surveys are badged and all exposure received during the surveys is included in the normal occupational exposure which is regularly monitored.

A.7.2.1.11 Traffic Count Passing a Specific Point. The RADTRAN 4 computer code uses 470 vehicles per hour passing the transport vehicle. Travel on the transport path is restricted to INEL employees by a security checkpoint, the majority of INEL employees ride the INEL site buses to work, and the transfers are not made during high traffic times (i.e., shift changes when buses are in service); therefore, using the standard 470 vehicles per hour value would be extremely conservative. A more realistic estimate of 25 vehicles per hour was used.

A.7.2.1.12 Standard RADTRAN 4 Computer Code Values Used. The following standard RADTRAN 4 computer code value was reviewed and determined to reflect the best estimate of current industry practice and was consistent with historical data from transfers of naval spent nuclear fuel to storage:

- Minimum Number of Inspections.

The following standard RADTRAN 4 estimate of the population that could be affected by the transfer of naval spent nuclear fuel to storage was used to evaluate the five alternatives:

- Number of People per Vehicle Sharing the Transport Route (On Link).

A.7.2.2 Accident During Transportation of Spent Nuclear Fuel to Storage. This section provides the input parameters used to calculate the radiological impacts for accidents during transportation of spent nuclear fuel to storage for evaluation of the five alternatives. The planned transfers, transportation distances, population densities, and the percentages of travel in each population density described in Section A.7.2.1 were also used for the accident analyses. Unless otherwise described in this section, the standard values provided by the RADTRAN 4 and RISKIND computer codes were used.

A.7.2.2.1 Accident Probability. The probability of a truck accident used for evaluation of all alternatives was obtained from "Longitudinal Review of State-Level Accident Statistics for Carriers of Interstate Freight" (Saricks and Kvitek 1994). The truck accident rates are state dependent. The states in which naval spent nuclear fuel would be transferred to storage for the alternatives described in Section A.3 are Idaho, Washington, South Carolina, Tennessee, and Nevada. The corresponding accident rates for travel on rural interstates in accidents per kilometer are 2.30×10^{-7} for Idaho, 2.50×10^{-7} for Washington, 1.83×10^{-7} for South Carolina, 1.48×10^{-7} for Tennessee, and 1.57×10^{-7} for Nevada. The values correspond to 3.70×10^{-7} (Idaho), 4.02×10^{-7} (Washington), 2.94×10^{-7} (South Carolina), 2.38×10^{-7} (Tennessee), and 2.53×10^{-7} (Nevada) accidents per mile.

A.7.2.2.2 Accident Severity Categories and Probabilities. In the modal study, Lawrence Livermore National Laboratory categorized the potential damage to shipping containers according to the magnitude of the thermal and mechanical forces that could result from an accident. The structural and thermal forces were categorized into 20 regions. Given that an accident occurs, the probability that the accident would be in each region was calculated for both rail and truck shipments. Table A-17 provides the probabilities for truck accidents by region.

Table A-17. Accident severity probabilities for truck shipments.

Structural Response (maximum strain on inner shell, %)					
	R(4,1) 1.532×10^{-7}	R(4,2) 3.926×10^{-14}	R(4,3) 1.495×10^{-14}	R(4,4) 7.681×10^{-16}	R(4,5) $<1 \times 10^{-16}$
S_3 (30)	R(3,1) 1.7984×10^{-3}	R(3,2) 1.574×10^{-7}	R(3,3) 2.034×10^{-7}	R(3,4) 1.076×10^{-7}	R(3,5) 4.873×10^{-8}
S_2 (2)	R(2,1) 3.8192×10^{-3}	R(2,2) 2.330×10^{-7}	R(2,3) 3.008×10^{-7}	R(2,4) 1.592×10^{-7}	R(2,5) 7.201×10^{-8}
S_1 (0.2)	R(1,1) 0.994316	R(1,2) 1.687×10^{-5}	R(1,3) 2.362×10^{-5}	R(1,4) 1.525×10^{-5}	R(1,5) 9.570×10^{-6}
	T_1 (500)	T_2 (600)	T_3 (650)	T_4 (1050)	

Thermal Response (lead mid-thickness temperature, °F)

A.7.2.2.3 Naval Spent Nuclear Fuel Integrity Following an Accident. Detailed structural and thermal analyses have been performed for the casks used for shipments of naval spent nuclear fuel to storage. As described in Section A.4.5, these analyses are performed using a worst credible accident which is defined based on the site specific terrain and administrative controls during the short on-site shipment. The probability of the worst credible accident is equal to that listed in region R(1,1). For accident conditions in excess of the worst credible accident, the fission product and corrosion product release fractions described in the next section were used.

A.7.2.2.4 Cask Release Fractions. The cask release fractions were derived based on the results presented in the NRC modal study (NUREG 1987). Although the naval spent nuclear fuel is stronger, the release fractions for the BWR, PWR, and aluminum-clad fuel from the modal study were used. From the modal study, the release fraction for lower left region R(1,1) is zero for the risk evaluation. For the maximum consequence evaluation, 1% of the corrosion products were released for the lower left region, R(1,1). The remaining regions used 10% of the fission products available for release, released at the fractions specified below, and release of 10% of the corrosion products. Table A-18 provides the release fractions used. The release fractions in Table A-18 for the less severe conditions differ from those in Table A-13 because supplementary structural and thermal analyses have not been performed for the casks discussed in this section.

Table A-18. Cask release fractions used for the RADTRAN 4 risk analyses.

Cask Response Region	Release Fraction[a] Inert Gas	Iodine	Cesium	Ruthenium	Particulates	Corrosion Products
R(1,1)	0.0	0.0	0.0	0.0	0.0	0.0
R(1,2), R(1,3)	9.9×10^{-3}	7.5×10^{-5}	6.0×10^{-6}	8.1×10^{-7}	6.0×10^{-8}	1.0
R(2,1), R(2,2), R(2,3)	3.3×10^{-2}	2.5×10^{-4}	2.0×10^{-5}	2.7×10^{-6}	2.0×10^{-7}	1.0
R(1,4), R(2,4), R(3,4)	3.9×10^{-1}	4.3×10^{-3}	2.0×10^{-4}	4.8×10^{-5}	2.0×10^{-6}	1.0
R(3,1), R(3,2), R(3,3)	3.3×10^{-1}	2.5×10^{-3}	2.0×10^{-4}	2.7×10^{-5}	2.0×10^{-6}	1.0
R(1,5), R(2,5), R(3,5) R(4,5), R(4,1), R(4,2) R(4,3), R(4,4)	6.3×10^{-1}	4.3×10^{-2}	2.0×10^{-3}	4.3×10^{-4}	2.0×10^{-5}	1.0

[a] The release fraction represents the fraction of the fuel inventory available for release in the cask that would be released into the atmosphere following an accident of the given severity.

A.7.2.2.5 Plume Release Height. For the accident risk assessment, a ground level release was used. For the maximum consequence assessment, a plume release height of 10 meters (32.8 feet) was used.

A.7.2.2.6 Direct Exposure from a Damaged Shipping Container. A radiation level following the accident at the 10CFR71 regulatory limit of 1 rem at 1 meter (3.3 feet) from the cask surface was used.

A.7.2.2.7 Food Transfer Factors. Food transfer factors were derived for the isotopes related to naval spent nuclear fuel in accordance with the methods described in Nuclear Regulatory Commission Guide 1.109 (NUREG 1977).

A.7.2.2.8 Distance from the Accident Scene to the Maximum Exposed Individual. No shielding was accounted for as the plume passes for the calculation of the exposure to the maximum individual. This location was determined using RISKIND based on the selected atmospheric stability and plume release height. The maximum exposed individual could be a member of the track crew or the general population.

A.7.2.2.9 RISKIND Population Density. From the HIGHWAY computer code, the population density for the on-site shipment was determined to be one person per square kilometer (2.6 persons per square mile) in a rural area. For on-site transportation at INEL, the population density in the most populated sector, from 1990 census data, is 55 people per square kilometer, with the majority of these people in the area 64.4 to 80 kilometers (40 to 50 miles) from the site. This population density is just into the lower region of the suburban density range of 53.7 to 1284.7 people per square kilometer (139 to 3326 people per square mile) used in HIGHWAY and INTERLINE. The standard value of 6 (rural) and 719 (suburban) people per square kilometer (15.5 and 1861 people per square mile, respectively) was used for the evaluation of all alternatives.

A.7.2.2.10 Radionuclide Inventory. The transfers of naval spent nuclear fuel to storage contain the same radionuclides as listed in Table A-14. On average, there is approximately 80 percent of the activity of each radionuclide.

A.7.3 Transfers of Naval Test Specimen Assemblies Between the Examination Facility and the Test Reactor Area

A.7.3.1 Incident-free Transportation of Naval Test Specimen Assemblies. This section provides the input parameters used to determine the radiological impacts associated with the routine, incident-free (i.e., no accident) transportation of naval test specimen assemblies for each of the five alternatives.

A.7.3.1.1 Planned Shipments. Table A-19 provides the number of planned transfers in each cask.

Table A-19. Planned transfers of naval test specimen assemblies.

	NR/ATR	Test Train
No Action, Decentralization - No Exam, Decentralization - Limited Exam	0	0
Decentralization - Full Exam, 1992/1993 Planning Basis, Regionalization at INEL, and Centralization at INEL	38	922
All other Regionalization and Centralization Alternatives	0	960

A.7.3.1.2 Transport Index. A TI of 130.0 was used for all NR and ATR cask transfers. This value was derived from historic measurements over the last several years. The new Test Train casks, which are currently being designed, would have a TI of 1.0.

A.7.3.1.3 Transportation Distances and Population Densities. Section A.6 provided a description of the general methodology used for determining transportation distances and the population densities along the transportation routes. The distance between ECF and TRA is 8.0 kilometers (5 miles). From the HIGHWAY computer code, this on-site transfer of naval test specimen assemblies occurs in a rural area. For shipments from TRA to the centralization sites, the HIGHWAY computer code was used to calculate the distance traveled, the population densities, and the percent distance traveled in each population density. As described in Section A.7.4.1.3, the HIGHWAY predicted distances for off-site shipments were increased by 3%.

A.7.3.1.4 Truck Speed. The standard RADTRAN 4 computer code speed for truck shipments in a rural population is 88.5 kilometers per hour (55 miles per hour). One of the reasons an on-site worst credible accident is less severe than the 10CFR71 hypothetical accident is that the speed is severely limited. An average speed of 16.1 kilometers per hour (10 miles per hour) was used for the on-site shipments. For off-site shipments to the centralization sites, the standard RADTRAN 4 computer code values were used.

A.7.3.1.5 Truck Stop Time. The standard RADTRAN 4 computer code provides values for truck stop times that are either dependent or independent of the distances traveled. The logs for historical on-site transfers of naval test specimen assemblies were reviewed, and it was determined that the actual stop time (one and one-half hours) was less than the standard RADTRAN 4 computer code values. For the alternative in which on-site transfers would continue, the one and one-half hour stop time was used. For the off-site shipments of test specimen assemblies to the centralization sites, a stop time of 0.006 hour per kilometer (0.01 hour per mile) was used, consistent with the value used for other past truck shipments outside the boundaries of DOE facilities (see Section A.7.4.1.4).

A.7.3.1.6 Radiation Exposure Decrease Due to Distance. The radiation exposure decrease due to distance described in Section A.7.1.1.9 was also applied to the truck transfers of test specimen assemblies.

A.7.3.1.7 Distance from Source to Crew. A distance of 3.6 meters (12 feet) was measured between the NR/ATR shipping cask and the driver for the exclusive-use truck transfers of test specimen assemblies on-site. Two escorts, one located approximately 46 meters (150 feet) in front and one the same distance behind the transport vehicle, are also present for on-site shipments.

For off-site shipments to the centralization sites, the standard RADTRAN 4 computer code value for the number of crew members was used (2). The value used for the distance from the crew to the centerline of the cask for off-site shipments was 5.85 meters (20 feet), based on the conceptual design of the new Test Train cask.

A.7.3.1.8 Transport Index to Exposure Rate Conversion Factors. Transport index to exposure rate conversion factors for the casks used for test specimen assembly transfers were calculated using the standard equation used by RADTRAN 4. The results were compared to detailed computer analyses performed using SPAN4, and RADTRAN 4 results were found to overestimate the exposure. Using the SPAN4 computer code results, the effective package dimensions of the casks used in the RADTRAN 4 calculations were adjusted to provide a conservative yet more realistic value of the transport index to exposure rate conversion factor. The values used are provided in Table A-20.

Table A-20. Transport index to exposure rate conversion factors for the NR/ATR and Test Train casks.

Cask	Effective Package Dimension (meters)	Transport Index to Exposure Rate Conversion Factor
NR/ATR	0.61 (2 feet)	1.70
Test Train	1.70 (5.6 feet)	3.42

A.7.3.1.9 Storage. There is no intermediate storage time during transfers of naval test specimen assemblies.

A.7.3.1.10 Persons Exposed While Stopped. The only stop time for the transfer of naval test specimen assemblies on-site occurs during routine surveys at the destination entrance. This area is well removed from highway and population and therefore no people were considered to be exposed during the one and one-half hour stop. The escorts are not present during the surveys and the driver is positioned approximately 46 meters (150 feet) from the source during the surveys. The people performing the surveys are badged and all exposure received during the survey is included in the normal occupational exposure which is regularly monitored. For off-site shipments, the standard RADTRAN 4 computer code values were used.

A.7.3.1.11 Traffic Count Passing a Specific Point. The RADTRAN 4 computer code uses 470 vehicles per hour passing the transport vehicle. Travel on the on-site transport path is restricted to INEL employees, the majority of INEL employees ride the INEL site buses to work, and the transfers are not made during high traffic times (i.e., shift changes); therefore, using the standard 470 vehicles per hour value would excessively overestimate the number of persons involved. A more realistic estimate of 25 vehicles per hour was used for on-site shipments. For off-site shipments, the standard RADTRAN 4 computer code values were used.

A.7.3.1.12 Standard RADTRAN 4 Computer Code Values Used. The following standard RADTRAN 4 computer code value was reviewed and determined to reflect the best estimate of current industry practice and was consistent with recorded data from transfers of naval test specimen assemblies:

- Minimum Number of Inspections.

The following standard RADTRAN 4 estimate of the population that could be affected by the transfer of test specimen assemblies was used for evaluation of the five alternatives:

- Number of People per Vehicle Sharing the Transport Route (On Link).

A.7.3.2 Accident During Transportation of Naval Test Specimen Assemblies. This section provides the input parameters used to calculate the radiological impacts for accidents during transportation of naval test specimen assemblies for evaluation of the five alternatives. The planned transfers, transportation distances, population densities, and the percentages of travel in each population density described in Section A.7.3.1 were also used for the accident analyses. Unless otherwise described in this section, the standard values provided by the RADTRAN 4 and RISKIND computer codes were used. All variables described in Section A.7.2.2 are applicable to these transfers with the exception of the RISKIND population density.

A.7.3.2.1 RISKIND Population Densities. For the Decentralization, 1992/1993 Planning Basis, Regionalization at INEL, and Centralization at INEL alternatives, the test specimen assembly transfers would occur on the INEL site. For these transfers, the same conditions described in Section A.7.2.2.9 were used. For the other Regionalization and Centralization alternative risk assessments, the population densities from RADTRAN 4 were used.

A.7.3.2.2 Release Fractions. For the Decentralization, 1992/1993 Planning Basis, and Regionalization at INEL, and Centralization at INEL alternatives, the test specimen assembly transfers would occur on the INEL site. For these transfers, the same conditions described in Sections A.7.2.2.3 and A.7.2.2.4 were used. For the other Regionalization and Centralization alternatives, the conditions described in Sections A.7.1.2.3 and A.7.1.2.4 were used.

A.7.3.2.3 Radionuclide Inventory. The radionuclides which would be released from an average transfer are listed in Table A-21, along with the activity. The values factor in the damage fractions and release fractions described in Section A.7.3.2.2. The radionuclides listed result in 99 percent of the exposure in each pathway.

Table A-21. Radionuclides which would be released from an average transfer of test specimen assemblies.

For Accidents which Release Both Fission and Corrosion Products		For Accidents which Release Only Corrosion Products	
Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
I-131	1.30×10^3	Eu-156	3.75×10^1
H-3	3.51×10^2	Lu-177	1.59×10^1
I-132	3.10×10^2	Eu-152	1.41×10^1
Eu-156	3.75×10^1	Zr-95	1.07×10^1
Eu-152	1.41×10^1	Zn-65	9.80×10^0
Zr-95	1.09×10^1	Co-60	7.68×10^0
Zn-65	9.80×10^0	Ce-141	6.60×10^0
Co-60	7.68×10^0	Eu-154	6.15×10^0
Eu-154	6.15×10^0	Cs-136	4.69×10^0
Sc-46	3.25×10^0	Sc-46	3.25×10^0
Cs-137	1.78×10^0	I-131	2.37×10^0
Ru-106	3.36×10^{-1}	Hf-181	2.35×10^0
Nb-95	2.64×10^{-1}		
Pr-144	2.19×10^{-1}		
Ce-144	2.19×10^{-1}		

A.7.4 Shipments of Naval Irradiated Test Specimens to Examination and Testing Facilities

A.7.4.1 Incident-free Transportation of Test Specimens. This section provides the input parameters used to determine the radiological impacts associated with the routine, incident-free (i.e., no accident) transportation of test specimens for evaluation of the five alternatives.

A.7.4.1.1 Planned Shipments. Table A-22 provides the estimated number of shipments used in the analysis.

Table A-22. Planned shipments of naval test specimens.

	NRBK-41/WAPD-40				
Alternative	ICPP	PSNS	Centralization Site	BETTIS	KAPL
No Action Decentralization - No Exam	29	0	0	0	320
Decentralization - Limited Exam	26	3	0	0	320
Decentralization - Full Exam	0	0	0	0	320
1992/1993 Planning Basis, Regionalization at INEL, and Centralization at INEL Alternatives	0	0	0	120	641
All other Regionalization and Centralization Alternatives	0	0	29	120	641

A.7.4.1.2 Transport Index. A TI of 0.1 was used for all NRBK-41 and WAPD-40 shipping container shipments. These values were derived from recorded measurements over the last several years.

A.7.4.1.3 Transportation Distances and Population Densities. Section A.6 provided a description of the general methodology used for determining transportation distances and the population densities along the transportation routes. Historical data were obtained for shipments of test specimens. The distance traveled was averaged based on the point of origin and compared to the value calculated by HIGHWAY. The actual distance traveled was approximately 3% higher on the average. In order to provide the best estimate exposure, which is based on the distance traveled, the HIGHWAY distances were increased by 3% for all alternatives. This technique allowed for comparison of the alternatives on an equal basis. The percentages of distance traveled in each population density calculated by HIGHWAY applied to the distances which were increased by the 3%.

A.7.4.1.4 Truck Stop Time. The RADTRAN 4 computer code provides standard values for truck stop times that are either dependent or independent of the distances traveled. The shipping logs for historical test specimen shipments were reviewed, and it was determined that the actual stop times were much shorter than the standard RADTRAN 4 computer code values. The recorded stop times were divided by the actual distance traveled from historical data over the last three years and an average of 0.006 hour per kilometer (0.01 hour per mile) was calculated. This value was used to evaluate all five alternatives.

A.7.4.1.5 Radiation Exposure Decrease Due to Distance. The radiation exposure decrease due to distance described in Section A.7.1.1.9 was also applied to the truck shipments of test specimens.

A.7.4.1.6 Transport Index to Exposure Rate Conversion Factors. Container transport index to exposure rate conversion factors for the casks used for test specimen shipments were calculated using the standard equation used by RADTRAN 4. The results were compared to detailed computer analyses performed using SPAN4, and RADTRAN 4 results were found to overestimate the exposure. Using the SPAN4 computer code results, the effective package dimensions of the containers used in the RADTRAN 4 calculations were adjusted to provide a conservative yet more realistic value of the transport index to exposure rate conversion factor. The values used are provided in Table A-23.

Table A-23. Transport index to exposure rate conversion factors for the NRBK-41 and WAPD-40 shipping containers.

Container	Effective Package Dimension (meters)	Transport Index to Exposure Rate Conversion Factor
NRBK-41	0.74 (2.4 feet)	1.88
WAPD-40	3.2 (10.5 feet)	6.76

A.7.4.1.7 Storage. The test specimen shipping containers are not stored during shipment.

A.7.4.1.8 Standard RADTRAN 4 Computer Code Values Used. The following standard RADTRAN 4 computer code values were reviewed and were determined to reflect the best estimate of current industry practice and were consistent with historical data from shipments of naval test specimens:

- Truck Speed
- Distance from Source to Crew
- Number of Crewmen
- Minimum Number of Inspections.

The following standard RADTRAN 4 estimates of the populations that could be affected by the shipment of test specimens were also used to evaluate the five alternatives:

- Persons Exposed While Stopped
- Average Exposure Distance While Stopped
- Number of People per Vehicle Sharing the Transport Route (On Link)
- Traffic Count Passing a Specific Point - Rural, Suburban, and Urban Zones
- Fraction of Travel During Rush Hour, on City Streets, and on Freeways.

A.7.4.2 Accident During Transportation of Test Specimens. This section provides the input parameters used to calculate the radiological impacts for accidents during transportation of test specimens to evaluate the five alternatives. The planned shipments, transportation distances, population densities, and the percentages of travel in each population density described in Section A.7.4.1 were also used for the accident analyses. Unless otherwise described in this section, the standard values provided by the RADTRAN 4 and RISKIND computer codes were used. All the conditions and variables described in Section A.7.1.2 are applicable to these shipments with the exception of the Accident Probability.

A.7.4.2.1 Accident Probability. The probability of a truck accident used for evaluation of all alternatives was obtained from "Longitudinal Review of State-Level Accident Statistics for Carriers of Interstate Freight" (Saricks and Kvitek 1994). The truck accident rates are state dependent. The states in which naval spent nuclear fuel would be shipped to storage for the alternatives described in

Section A.3 were obtained from HIGHWAY. The accident rate values are consistent with past test specimen shipments which have traveled approximately 2.4 million kilometers (1.5 million miles) without an accident.

A.7.4.2.2 Test Specimen Integrity Following an Accident. Detailed structural and thermal analyses were performed for the shipping containers used for naval test specimen shipments up to an equivalent strain of 30% and mid-wall temperature of 1050°F. For these cases, the sealed inner container was not damaged; therefore, only the activity on the outside of the inner container, which would be corrosion products, was released. For the thermal and structural regions above 1050°F and 30% strain, the modal study defines the upper limits as unbounded. For these cases, the sealed inner container holding the test specimens was postulated to be damaged and the fission products and corrosion products would be released in the quantities described in Section A.7.1.2.4.

A.7.4.2.3 Radionuclide Inventory. The test specimen shipments contain the same radionuclides as listed in Table A-21. On average, there is approximately 1.5 percent of the activity of each nuclide.

A.8 SUMMARY OF RESULTS

A.8.1 Historical - Incident Free

This section summarizes the results of the calculations for the radiological and non-radiological impacts of the incident-free transportation of naval spent nuclear fuel and test specimens. Table A-24 shows the radiological impact on the general population, transportation workers (occupational), and the maximum exposed individual, and the non-radiological impact on all persons. The radiological impact on the general population for all historical shipments is 1.95 person-rem, which statistically corresponds to 0.00098 cancer fatalities in the entire population over the 40-year period considered. The radiological impact on transportation workers for all historical shipments is 16.6 person-rem, which statistically corresponds to 0.0066 cancer fatalities. As can be seen from Table A-24, the radiological impact to the general population is greatest for the highway transportation of test specimens. Incident-free radiological impacts tend to be greater for highway transportation than for rail transportation since both the general population and transportation workers are closer to the shipping container in transit. In all cases, the maximum exposed individual is a

Table A-24. Incident-free results for historical Navy shipments.

	General Population		Occupational		MEI-General Population		MEI-Occupational		Estimated Non-Radiological Fatalities
	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Dose (rem)	Estimated Cancer Fatalities	Dose (rem)	Estimated Cancer Fatalities	
Naval Spent Nuclear Fuel to ECF[2]	0.70	3.5×10^{-4}	3.2	1.3×10^{-3}	0.033	1.7×10^{-5}	0.10	4.0×10^{-5}	1.6×10^{-2}
Naval Spent Nuclear Fuel to ICPP[1]	0.10	5.0×10^{-5}	2.8	1.1×10^{-3}	2.1×10^{-5}	1.1×10^{-8}	2.8	1.1×10^{-3}	0
Test Specimen Assemblies Between ECF and TRA[1]	0.22	1.1×10^{-4}	7.6	3.0×10^{-3}	0.062	3.1×10^{-5}	7.5	3.0×10^{-3}	0
Test Specimens[2]	0.93	4.7×10^{-4}	3.0	1.2×10^{-3}	0.026	1.3×10^{-5}	1.5	6.0×10^{-4}	1.2×10^{-2}
TOTAL[3]	1.95	9.8×10^{-4}	16.6	6.6×10^{-3}	0.062	3.1×10^{-5}	7.5	3.0×10^{-3}	2.8×10^{-2}

[1] On-site
[2] Off-site
[3] Maximum Exposed Individual exposures are not cumulative, they are the maximum value.

73

transportation worker, since the workers are closer to the shipment for a longer time than any member of the general population. The maximum exposed individual for all shipments is a driver for the trucks transferring test specimen assemblies between ECF and TRA. Under the limiting modeling approach that the same person drove every shipment for the entire period, this person received a total exposure of 7.5 rem over the approximate 40-year period, or about 0.19 rem per year, which is within DOE limits for occupationally exposed individuals. By comparison, the maximum exposed individual for the general population received only 0.062 rem over the entire historical period, which is much less than the exposure to the maximum exposed individual transportation worker and corresponds to 0.0016 mrem exposure per year. It should be noted that the majority of the exposure to the transportation worker and maximum exposed worker is already accounted for since most transportation workers are badged and therefore this exposure is included with all other exposure they would receive on the job. The rail employees and off-site truck drivers are the only transportation workers who are not badged. Their exposure was calculated to be only approximately 30% of the total.

The estimated non-radiological fatalities due to vehicle emissions is 0.028 for the entire 40-year period.

A.8.2 Incident Free

Table A-25 provides a summary of the annual exposures and risks from incident-free transportation of naval spent nuclear fuel and test specimens for all alternatives. The values are calculated by dividing the values in Table A-26 by the 40 years evaluated to obtain the average annual values.

The annual radiological impact on the general population ranges from 0.0085 to 0.30 person-rem. The general population annual radiological risk ranges from 0.0000043 to 0.0C015 for cancer fatalities.

The radiological impact on the transportation crew (occupational) ranges from 0.038 to 0.38 person-rem. The transportation crew annual radiological risk ranges from 0.000015 to 0.00015 for cancer fatalities.

Table A-25. Summary of annual incident-free impacts during transportation of naval spent nuclear fuel and test specimens.

	General Population		Occupational		MEI-General Population		MEI-Occupational		Estimated Non-Radiological Fatalities (per year)
	Collective Dose (person-rem/yr)	Estimated Cancer Fatalities (per year)	Collective Dose (person-rem/yr)	Estimated Cancer Fatalities (per year)	Dose (rem/yr)	Estimated Cancer Fatalities (per year)	Dose (rem/yr)	Estimated Cancer Fatalities (per year)	
No Action	0.0085	4.3×10^{-6}	0.038	1.5×10^{-5}	0.00098	4.9×10^{-7}	0.0087	3.5×10^{-6}	1.5×10^{-4}
Decentralization - No Exam	0.0085	4.3×10^{-6}	0.038	1.5×10^{-5}	0.00098	4.9×10^{-7}	0.0087	3.5×10^{-6}	1.5×10^{-4}
Decentralization - Limited Exam	0.021	1.1×10^{-5}	0.068	2.7×10^{-5}	0.0011	5.5×10^{-7}	0.0087	3.5×10^{-6}	2.2×10^{-4}
Decentralization - Full Exam	0.083	4.2×10^{-5}	0.30	1.2×10^{-4}	0.0043	2.2×10^{-6}	0.032	1.3×10^{-5}	7.5×10^{-4}
1992-1993 Planning Basis	0.053	2.7×10^{-5}	0.18	7.2×10^{-5}	0.0022	1.1×10^{-6}	0.020	8.0×10^{-6}	6.3×10^{-4}
Regionalization or Centralization at INEL	0.053	2.7×10^{-5}	0.18	7.2×10^{-5}	0.0022	1.1×10^{-6}	0.020	8.0×10^{-6}	6.3×10^{-4}
Regionalization or Centralization at Hanford	0.12	6.0×10^{-5}	0.25	1.0×10^{-4}	0.0040	2.0×10^{-6}	0.027	1.1×10^{-5}	8.8×10^{-4}
Regionalization or Centralization at Savannah River	0.30	1.5×10^{-4}	0.38	1.5×10^{-4}	0.0040	2.0×10^{-6}	0.12	4.8×10^{-5}	8.3×10^{-4}
Regionalization or Centralization at Oak Ridge	0.28	1.4×10^{-4}	0.35	1.4×10^{-4}	0.0040	2.0×10^{-6}	0.10	4.0×10^{-5}	7.0×10^{-4}
Regionalization or Centralization at Nevada Test Site	0.15	7.5×10^{-5}	0.28	1.1×10^{-4}	0.0040	2.0×10^{-6}	0.042	1.7×10^{-5}	9.3×10^{-4}

75

Table A-26. Summary of 40-year cumulative incident-free impacts during transportation of naval spent nuclear fuel and test specimens.

	General Population		Occupational		MEI-General Population		MEI-Occupational		Estimated Non-Radiological Fatalities
	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Dose (rem)	Estimated Cancer Fatalities	Dose (rem)	Estimated Cancer Fatalities	
No Action	0.34	1.7×10^{-4}	1.5	6.0×10^{-4}	0.039	2.0×10^{-5}	0.35	1.4×10^{-4}	5.9×10^{-3}
Decentralization - No Exam	0.34	1.7×10^{-4}	1.5	6.0×10^{-4}	0.039	2.0×10^{-5}	0.35	1.4×10^{-4}	5.9×10^{-3}
Decentralization - Limited Exam	0.83	4.2×10^{-4}	2.7	1.1×10^{-3}	0.045	2.3×10^{-5}	0.35	1.4×10^{-4}	8.9×10^{-3}
Decentralization - Full Exam	3.3	1.7×10^{-3}	12	4.8×10^{-3}	0.17	[illegible].5 x 10^{-5}	0.43	1.7×10^{-4}	3.0×10^{-2}
1992-1993 Planning Basis	2.1	1.1×10^{-3}	7.3	2.9×10^{-3}	0.086	4.3×10^{-5}	0.80	3.2×10^{-4}	2.5×10^{-2}
Regionalization or Centralization at INEL	2.1	1.1×10^{-3}	7.3	2.9×10^{-3}	0.086	4.3×10^{-5}	0.80	3.2×10^{-4}	2.5×10^{-2}
Regionalization or Centralization at Hanford	4.7	2.4×10^{-3}	9.8	3.9×10^{-3}	0.16	8.0×10^{-5}	1.1	4.4×10^{-4}	3.5×10^{-2}
Regionalization or Centralization at Savannah River	12	6.0×10^{-3}	15	6.0×10^{-3}	0.16	8.0×10^{-5}	4.7	1.9×10^{-3}	3.3×10^{-2}
Regionalization or Centralization at Oak Ridge	11	5.5×10^{-3}	14	5.6×10^{-3}	0.16	8.0×10^{-5}	4.1	1.6×10^{-3}	2.8×10^{-2}
Regionalization or Centralization at Nevada Test Site	6.0	3.0×10^{-3}	11	4.4×10^{-3}	0.16	8.0×10^{-5}	1.7	6.8×10^{-4}	3.7×10^{-2}

For all alternatives, the maximum exposed individual is a transportation worker who drives the truck shipments. The annual radiological impact on the maximum exposed individual ranges from 0.0087 to 0.12 rem. These values were calculated based on the modeling approach that for each of the categories of shipments described in Sections A.4.2 through A.4.4, the same person would drive all shipments. The maximum exposed individual annual radiological risk ranges from 0.0000035 to 0.000048 for cancer fatalities. The annual exposure to the maximum exposed individual of the general population ranges from 0.00098 to 0.0043 rem for the various alternatives. The estimated exposure and health effects to the maximum exposed individual for the general population correspond to approximately a factor of 10 less than those estimated for the transportation worker.

The annual non-radiological risk ranges from 0.00015 to 0.00093 fatalities.

The summary of exposures and risks from incident-free transportation of naval spent nuclear fuel and test specimens for all alternatives are included in Table A-26 for the 40-year period.

The radiological impact on the general population ranges from 0.34 to 12 person-rem. The general population radiological risk for the entire 40-year period ranges from 0.00017 to 0.006 for cancer fatalities.

The radiological impact on the transportation crew (occupational) ranges from 1.5 to 15 person-rem. The transportation crew radiological risk for the entire 40-year period ranges from 0.0006 to 0.006 for cancer fatalities.

For all alternatives, the maximum exposed individual is a transportation worker who drives the truck shipments. The radiological impact on the maximum exposed individual ranges from 0.35 to 4.7 rem. These values were calculated based on using the same driver for all shipments for each of the categories of shipments described in Sections A.4.2 through A.4.4. The maximum exposed individual radiological risk for the entire 40-year period, 1995 through 2035, ranges from 0.00014 to 0.0019 for cancer fatalities. The exposure to the maximum exposed individual of the general population ranges from 0.039 to 0.17 rem for the various alternatives. The estimated exposure and health effects to the maximum exposed individual for the general population correspond to approximately a factor of 10 less than those estimated for the transportation worker.

The non-radiological risk ranges from 0.0059 to 0.037 fatalities for the entire 40-year period.

There are appreciable differences in exposure to the general population, transportation crew, and the maximum exposed individual among the various alternatives. Part of these differences is due to the varying number of shipments. For example, for the Decentralization - Full Examination alternative, all shipments of naval spent nuclear fuel are shipped to the INEL and then returned to the shipyards and prototypes, thereby doubling the number of shipments. However, the single most important contributor to the differences among the alternatives is the shipment of test specimen assemblies. For the No Action, Decentralization - No Examination, and Decentralization - Limited Examination alternatives, there are no shipments; for the Decentralization - Full Examination, 1992/1993 Planning Basis, Regionalization at INEL, and Centralization at INEL alternatives, the exposure is minimal since the shipments remain on the INEL site. However, for the other Regionalization and Centralization alternatives, the test specimen assemblies would be shipped off-site between the INEL and the alternative sites. While the exposure rates on the casks are low, the number of shipments and the distances involved increase the radiological impact on the transportation crew and the general population.

Tables A-27 and A-28 provide the 40-year cumulative incident-free results separately for on-site and off-site shipments. For all alternatives, the shipments of naval spent nuclear fuel from shipyards and prototypes and shipments of naval irradiated test specimens are off-site. Likewise, the transfers of naval spent nuclear fuel to storage following examination are on-site for all alternatives. The transfers of naval test specimen assemblies are off-site for the Regionalization and Centralization alternatives at Hanford, Savannah River, Oak Ridge, and the Nevada Test Site, otherwise they would be on-site.

As described in Section 3.8 of the main body of this Appendix, all alternatives which do not make use of the existing Expended Core Facility at INEL would require a transition period while new facilities for examination and storage of naval spent nuclear fuel were developed. During the transition period, approximately 80 shipments from Navy sites to ECF would be needed. These shipments are not included explicitly in the detailed analyses; however, the appropriate number of shipments needed by each alternative during this period is explicitly included, so the range of environmental effects of these shipments is bounded. For example, the estimated fatalities for the No Action, Decentralization - No Examination, and Decentralization - Limited Examination alternatives would actually increase slightly if the transition shipments were included. The estimated fatalities for the alternatives in which the INEL continues to receive shipments would remain the same. For the Regionalization and Centralization alternatives at sites other than INEL, the estimated fatalities would

Table A-27. Summary of 40-year cumulative incident-free impacts of on-site transportation.

	General Population		Occupational		MEI-General Population		MEI-Occupational		Estimated Non-Radiological Fatalities
	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Dose (rem)	Estimated Cancer Fatalities	Dose (rem)	Estimated Cancer Fatalities	
No Action	0.00010	5.0×10^{-8}	0.0018	7.2×10^{-7}	0.000017	8.5×10^{-9}	0.0017	6.8×10^{-7}	0
Decentralization - No Exam	0.00010	5.0×10^{-8}	0.0018	7.2×10^{-7}	0.000017	8.5×10^{-9}	0.0017	6.8×10^{-7}	0
Decentralization - Limited Exam	0.00010	5.0×10^{-8}	0.0018	7.2×10^{-7}	0.000017	8.5×10^{-9}	0.0017	6.8×10^{-7}	0
Decentralization - Full Exam	0.013	6.5×10^{-6}	0.44	1.8×10^{-4}	0.062	3.1×10^{-5}	0.43	1.7×10^{-4}	0
1992-1993 Planning Basis	0.015	7.5×10^{-6}	0.50	2.0×10^{-4}	0.062	3.1×10^{-5}	0.43	1.7×10^{-4}	0
Regionalization or Centralization at INEL	0.015	7.5×10^{-6}	0.50	2.0×10^{-4}	0.062	3.1×10^{-5}	0.43	1.7×10^{-4}	0
Regionalization or Centralization at Hanford	0.0024	1.2×10^{-6}	0.067	2.7×10^{-5}	0.000017	8.5×10^{-9}	0.065	2.6×10^{-5}	0
Regionalization or Centralization at Savannah River	0.0024	1.2×10^{-6}	0.067	2.7×10^{-5}	0.000017	8.5×10^{-9}	0.065	2.6×10^{-5}	0
Regionalization or Centralization at Oak Ridge	0.0024	1.2×10^{-6}	0.067	2.7×10^{-5}	0.000017	8.5×10^{-9}	0.065	2.6×10^{-5}	0
Regionalization or Centralization at Nevada Test Site	0.0024	1.2×10^{-6}	0.067	2.7×10^{-5}	0.000017	8.5×10^{-9}	0.065	2.6×10^{-5}	0

79

Table A-28. Summary of 40-year cumulative incident-free impacts of off-site transportation.

	General Population		Occupational		MEI-General Population		MEI-Occupational		Estimated Non-Radiological Fatalities
	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Dose (rem)	Estimated Cancer Fatalities	Dose (rem)	Estimated Cancer Fatalities	
No Action	0.34	1.7×10^{-4}	1.5	6.0×10^{-4}	0.039	2.0×10^{-5}	0.35	1.4×10^{-4}	5.9×10^{-3}
Decentralization - No Exam	0.34	1.7×10^{-4}	1.5	6.0×10^{-4}	0.039	2.0×10^{-5}	0.35	1.4×10^{-4}	5.9×10^{-3}
Decentralization - Limited Exam	0.83	4.2×10^{-4}	2.7	1.1×10^{-3}	0.045	2.3×10^{-5}	0.35	1.4×10^{-4}	8.9×10^{-3}
Decentralization - Full Exam	3.3	1.7×10^{-3}	11	4.4×10^{-3}	0.17	8.5×10^{-5}	0.35	1.4×10^{-4}	3.0×10^{-2}
1992-1993 Planning Basis	2.1	1.1×10^{-3}	6.8	2.7×10^{-3}	0.086	4.3×10^{-5}	0.80	3.2×10^{-4}	2.5×10^{-2}
Regionalization or Centralization at INEL	2.1	1.1×10^{-3}	6.8	2.7×10^{-3}	0.086	4.3×10^{-5}	0.80	3.2×10^{-4}	2.5×10^{-2}
Regionalization or Centralization at Hanford	4.7	2.4×10^{-3}	9.7	3.9×10^{-3}	0.16	8.0×10^{-5}	1.1	4.4×10^{-4}	3.5×10^{-2}
Regionalization or Centralization at Savannah River	12	6.0×10^{-3}	15	6.0×10^{-3}	0.16	8.0×10^{-5}	4.7	1.9×10^{-3}	3.3×10^{-2}
Regionalization or Centralization at Oak Ridge	11	5.5×10^{-3}	14	5.6×10^{-3}	0.16	8.0×10^{-5}	4.1	1.6×10^{-3}	2.8×10^{-2}
Regionalization or Centralization at Nevada Test Site	6.0	3.0×10^{-3}	11	4.4×10^{-3}	0.16	8.0×10^{-5}	1.7	6.8×10^{-4}	3.7×10^{-2}

also remain approximately the same since the number of shipments is approximately evenly distributed between the east and west coast origins and therefore the total distance traveled is the same.

A.8.3 Accident Risk

This section summarizes the re ılts of the calculations for radiological and non-radiological risks from accidents which could occu uring shipments of naval spent nuclear fuel and test specimens. Tables A-29 and A-30 pr vide the results of the accident risk assessment for each alternative. The risks are provided for the general population in terms of exposure and estimated cancer fatalities. The risks are presented for 50% meteorological conditions, Pasquill Stability Class D. Table A-29 provides the risks on an annual basis and Table A-30 provides the total risks over the entire 40-year period.

The annual radiological impact, from Table A-29, on the general population ranges from 0.00021 to 0.021 person-rem. These exposures equate to 0.00000011 to 0.000011 estimated cancer fatalities. For non-radiological impacts, the estimated annual fatalities from traffic accidents range from 0.0012 to 0.022.

The cumulative radiological impact, from Table A-30, on the general population ranges from 0.0082 to 0.84 person-rem. These exposures equate to 0.0000041 to 0.00042 estimated cancer fatalities. For non-radiological impacts, the estimated fatalities from traffic accidents range from 0.047 to 0.84.

There are appreciable differences in exposure to the general population, transportation crew, and the maximum exposed individual among the various alternatives. Part of these differences is due to the varying number of shipments. For example, for the Decentralization - Full Examination alternative, all shipments of naval spent nuclear fuel are shipped to the INEL and then returned to the shipyards and prototypes, thereby doubling the number of shipments. As in the incident-free assessment, the shipment of test specimen assemblies is a large factor. For the No Action, Decentralization - No Examination, and Decentralization - Limited Examination alternatives, there are no shipments; for the Decentralization - Full Examination, 1992/1993 Planning Basis, Regionalization at INEL, and Centralization at INEL alternatives, the exposure is minimal since the shipments remain

Table A-29. Summary of annual accident risk for transportation of naval spent nuclear fuel and test specimens.

	General Population Collective Dose (person-rem/yr)	Estimated Cancer Fatalities (per year)	Estimated Traffic Fatalities (per year)
	Class D	Class D	
No Action	0.00021	1.1×10^{-7}	1.2×10^{-3}
Decentralization - No Exam	0.00021	1.1×10^{-7}	1.2×10^{-3}
Decentralization - Limited Exam	0.00043	2.2×10^{-7}	1.6×10^{-3}
Decentralization - Full Exam	0.0028	1.4×10^{-6}	2.2×10^{-2}
1992/1993 Planning Basis	0.0020	1.0×10^{-6}	1.3×10^{-2}
Regionalization or Centralization at INEL	0.0020	1.0×10^{-6}	1.3×10^{-2}
Regionalization or Centralization at Hanford	0.0033	1.7×10^{-6}	1.3×10^{-2}
Regionalization or Centralization at Savannah River	0.0210	1.1×10^{-5}	1.5×10^{-2}
Regionalization or Centralization at Oak Ridge	0.015	7.5×10^{-6}	1.4×10^{-2}
Regionalization or Centralization at Nevada Test Site	0.0070	3.5×10^{-6}	1.5×10^{-2}

Table A-30. Summary of cumulative accident risk over the 40-year period for transportation of naval spent nuclear fuel and test specimens.

	General Population Collective Dose (person-rem)	Estimated Cancer Fatalities	Estimated Traffic Fatalities
	Class D	Class D	
No Action	0.0082	4.1×10^{-6}	4.7×10^{-2}
Decentralization - No Exam	0.0082	4.1×10^{-6}	4.7×10^{-2}
Decentralization - Limited Exam	0.017	8.5×10^{-6}	6.5×10^{-2}
Decentralization - Full Exam	0.11	5.5×10^{-5}	8.6×10^{-1}
1992/1993 Planning Basis	0.079	4.0×10^{-5}	5.1×10^{-1}
Regionalization or Centralization at INEL	0.079	4.0×10^{-5}	5.1×10^{-1}
Regionalization or Centralization at Hanford	0.13	6.5×10^{-5}	5.3×10^{-1}
Regionalization or Centralization at Savannah River	0.84	4.2×10^{-4}	6.0×10^{-1}
Regionalization or Centralization at Oak Ridge	0.61	3.1×10^{-4}	5.7×10^{-1}
Regionalization or Centralization at Nevada Test Site	0.28	1.4×10^{-4}	6.1×10^{-1}

on the INEL site. However, for the other Regionalization and Centralization alternatives, the test specimen assemblies would be shipped off-site between the INEL and the alternate sites. While the exposure rates on the containers are low, the number of shipments and the distances involved increase the radiological impact on the transportation crew and the general population. In addition, the routes themselves are an important factor. While differences in distance and population densities are important, the higher risk for the Regionalization at Savannah River and Centralization at Savannah River alternatives, in particular, is due to the higher accident rates along the route taken and higher food transfer factors for shipments through farming states with much higher ingestion rates.

Table A-31 provides the 40-year cumulative risk, separated by on-site and off-site shipments.

As described in Section 3.8 of the main body of this Appendix, a transition period could be necessary which would require approximately 80 shipments from Navy sites to ECF. These shipments are not included explicitly in the detailed analyses; however, the appropriate number of shipments engendered by each alternative during this period is explicitly included, so the range of environmental effects of these shipments is bounded. The addition of the transition shipments would increase the distance traveled for the No Action, Decentralization - No Examination, and Decentralization - Limited Examination alternatives. Since the accident risk is proportional to the distance traveled, the risk would increase slightly for these alternatives, which were the lowest of all alternatives. All other alternatives would remain the same. Therefore, incorporating the transition period would actually reduce the difference between alternatives from the standpoint of transportation effects.

A.8.4 Accident Maximum Consequences

This section summarizes the results of the calculations of maximum consequences of accidents which could occur during shipments of naval spent nuclear fuel and test specimens. Tables A-32 and A-33 provide the results of the maximum consequence assessment for each alternative. The maximum consequences are provided for the general population by population area (rural, suburban, and urban) and the maximum exposed individual in terms of exposure. The members of the transportation crew may be the maximum exposed individual.

Table A-31. Summary of cumulative risk over the 40-year period for transportation of naval spent nuclear fuel and test specimens (on-site/off-site).

	ON-SITE			OFF-SITE		
	General Population			General Population		
	Collective Dose (person-rem)	Estimated Cancer Fatalities	Estimated Traffic Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Estimated Traffic Fatalities
No Action	1.3×10^{-6}	6.5×10^{-10}	6.8×10^{-6}	0.0082	4.1×10^{-6}	4.7×10^{-2}
Decentralization - No Exam	1.3×10^{-6}	6.5×10^{-10}	6.8×10^{-6}	0.0082	4.1×10^{-6}	4.7×10^{-2}
Decentralization - Limited Exam	1.3×10^{-6}	6.5×10^{-10}	6.8×10^{-6}	0.017	8.5×10^{-6}	6.3×10^{-2}
Decentralization - Full Exam	4.1×10^{-5}	2.1×10^{-8}	3.2×10^{-4}	0.11	5.5×10^{-5}	8.4×10^{-1}
1992-1993 Planning Basis	1.3×10^{-4}	6.5×10^{-8}	6.1×10^{-4}	0.079	4.0×10^{-5}	5.0×10^{-1}
Regionalization or Centralization at INEL	1.3×10^{-4}	6.5×10^{-8}	6.1×10^{-4}	0.079	4.0×10^{-5}	5.0×10^{-1}
Regionalization or Centralization at Hanford	8.7×10^{-5}	4.4×10^{-8}	2.1×10^{-4}	0.13	6.5×10^{-5}	5.3×10^{-1}
Regionalization or Centralization at Savannah River	8.7×10^{-5}	4.4×10^{-8}	3.6×10^{-4}	0.84	4.2×10^{-4}	5.9×10^{-1}
Regionalization or Centralization at Oak Ridge	8.7×10^{-5}	4.4×10^{-8}	2.3×10^{-4}	0.61	3.1×10^{-4}	5.7×10^{-1}
Regionalization or Centralization at Nevada Test Site	8.7×10^{-5}	4.4×10^{-8}	1.6×10^{-4}	0.28	1.4×10^{-4}	6.0×10^{-1}

85

Table A-32. Summary of maximum consequences (person-rem) of an accident (Design Basis).

	MAXIMUM CONSEQUENCES DESIGN BASIS (accident probability between 1 and 1 x 10^{-6})			
	Maximum Exposed Individual (rem)	Rural (person-rem)	Suburban (person-rem)	Urban (person-rem)
No Action	0.0034	0.51	4.3	13
Decentralization - No Exam	0.0034	0.51	4.3	13
Decentralization - Limited Exam	0.014	4.0	4.3	13
Decentralization - Full Exam	0.045	7.4	25	13
1992/1993 Planning Basis	0.045	7.4	25	13
Regionalization or Centralization at INEL	0.045	7.4	25	13
Regionalization or Centralization at Hanford	0.25	38	100	56
Regionalization or Centralization at Savannah River	0.25	38	320	560
Regionalization or Centralization at Oak Ridge	0.25	38	320	560
Regionalization or Centralization at Nevada Test Site	0.25	38	320	560

Table A-33. Summary of maximum consequences (person-rem) of an accident (Beyond Design Basis).

	MAXIMUM CONSEQUENCES BEYOND DESIGN BASIS (accident probability between 1 x 10^{-6} and 1 x 10^{-7})							
	Maximum Exposed Individual		Rural		Suburban		Urban	
	Estimated Dose (rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Estimated Collective Dose (person-rem)	Estimated Fatal Cancers	Estimated Collective Dose (person-rem)	Estimated Cancer Fatalities
No Action	0.014	7.0 x 10^{-6}	4.0	2.0 x 10^{-3}	25	1.3 x 10^{-2}	23	1.2 x 10^{-2}
Decentralization - No Exam	0.014	7.0 x 10^{-6}	4.0	2.0 x 10^{-3}	25	1.3 x 10^{-2}	23	1.2 x 10^{-2}
Decentralization - Limited Exam	0.045	2.3 x 10^{-5}	7.4	3.7 x 10^{-3}	25	1.3 x 10^{-2}	130	6.5 x 10^{-2}
Decentralization - Full Exam	1.8	9.0 x 10^{-4}	2700	1.4	3300	1.7	130	6.5 x 10^{-2}
1992/1993 Planning Basis	2.2	1.1 x 10^{-3}	3300	1.7	4100	2.1	130	6.5 x 10^{-2}
Regionalization or Centralization at INEL	2.2	1.1 x 10^{-3}	3300	1.7	4100	2.1	130	6.5 x 10^{-2}
Regionalization or Centralization at Hanford	2.2	1.1 x 10^{-3}	3300	1.7	4100	2.1	560	2.8 x 10^{-1}
Regionalization or Centralization at Savannah River	2.2	1.1 x 10^{-3}	3300	1.7	4100	2.1	1700	8.5 x 10^{-1}
Regionalization or Centralization at Oak Ridge	2.2	1.1 x 10^{-3}	3300	1.7	4100	2.1	1700	8.5 x 10^{-1}
Regionalization or Centralization at Nevada Test Site	2.2	1.1 x 10^{-3}	3300	1.7	4100	2.1	1700	8.5 x 10^{-1}

For design basis accidents, the calculated exposure to the general population ranges from 0.51 person-rem in a rural area to 560 person-rem in an urban area. The risk associated with these exposures ranges from 0.00026 to 0.28 cancer fatalities. The exposure to the maximum exposed individual ranges from 0.0034 rem to 0.25 rem. The risk to the maximum individual ranges from 0.0000017 to 0.00013 cancer fatalities.

For beyond design basis accidents, the exposure to the general population ranges from 4.0 person-rem in a rural area to 4100 person-rem in a suburban area (in this case, the probability of the accident of the same consequence in the urban area was less than 1×10^{-7}). The risk associated with these exposures ranges from 0.002 to 2.1 cancer fatalities. The exposure to the maximum exposed individual ranges from 0.014 rem to 2.2 rem. The risk to the maximum individual ranges from 0.000007 to 0.0011 cancer fatalities.

The shipments of naval spent nuclear fuel from shipyards and prototypes, transfers of naval spent nuclear fuel to storage, transfers of test specimen assemblies to the examination facility, and shipments of test specimens to test facilities were evaluated for the maximum consequences of an accident. Although the naval spent nuclear fuel shipments contain a higher amount of activity per shipment, there are cases where the test specimen shipment consequences are larger. The consequences are larger primarily due to the higher number of shipments which increases the probabilities such that a more severe consequence is evaluated.

Tables A-34 and A-35 provide the maximum consequences, separated by on-site and off-site shipments, respectively.

As described in Section 3.8 of the main body of this Appendix, a transition period could be necessary which would require approximately 80 shipments from Navy sites to ECF. These shipments are not included explicitly in the detailed analyses; however, the appropriate number of shipments engendered by each alternative during this period is explicitly included, so the range of environmental effects of these shipments is bounded. Since all alternatives ship the same basic fuel types, the maximum consequences are determined by the probability of the accident which is a function of the distance traveled. As described in Section A.8.3, only the No Action, Decentralization - No Examination, and Decentralization - Limited Examination alternatives, which have the lowest estimated maximum consequences, would increase the distance traveled if the

Table A-34. Summary of maximum consequences of an on-site accident (Beyond Design Basis).

	MEI		Rural		Suburban		Urban	
	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities
No Action	0.0013	6.5×10^{-7}	0.37	1.9×10^{-4}	2.4	1.2×10^{-3}	N/A	N/A
Decentralization - No Exam	0.0013	6.5×10^{-7}	0.37	1.9×10^{-4}	2.4	1.2×10^{-3}	N/A	N/A
Decentralization - Limited Exam	0.0013	6.5×10^{-7}	0.37	1.9×10^{-4}	2.4	1.2×10^{-3}	N/A	N/A
Decentralization - Full Exam	0.51	2.6×10^{-4}	200	1.0×10^{-1}	100	5.0×10^{-2}	N/A	N/A
1992-1993 Planning Basis	2.2	1.1×10^{-3}	3300	1.7	4100	2.1	N/A	N/A
Regionalization or Centralization at INEL	2.2	1.1×10^{-3}	3300	1.7	4100	2.1	N/A	N/A
Regionalization or Centralization at Hanford	2.2	1.1×10^{-3}	3300	1.7	4100	2.1	N/A	N/A
Regionalization or Centralization at Savannah River	2.2	1.1×10^{-3}	3300	1.7	4100	2.1	N/A	N/A
Regionalization or Centralization at Oak Ridge	2.2	1.1×10^{-3}	3300	1.7	4100	2.1	N/A	N/A
Regionalization or Centralization at Nevada Test Site	2.2	1.1×10^{-3}	3300	1.7	4100	2.1	N/A	N/A

Table A-35. Summary of maximum consequences of an off-site accident (Beyond Design Basis).

	MEI		Rural		Suburban		Urban	
	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities	Collective Dose (person-rem)	Estimated Cancer Fatalities
No Action	0.014	7.0×10^{-6}	4.0	2.0×10^{-3}	25	1.3×10^{-2}	23	1.2×10^{-2}
Decentralization - No Exam	0.014	7.0×10^{-6}	4.0	2.0×10^{-3}	25	1.3×10^{-2}	23	1.2×10^{-2}
Decentralization - Limited Exam	0.045	2.3×10^{-5}	7.4	3.7×10^{-3}	25	1.3×10^{-2}	130	6.5×10^{-2}
Decentralization - Full Exam	1.8	9.0×10^{-4}	2700	1.4	3300	1.7	130	6.5×10^{-2}
1992-1993 Planning Basis	1.8	9.0×10^{-4}	2700	1.4	79	4.0×10^{-2}	130	6.5×10^{-2}
Regionalization or Centralization at INEL	1.8	9.0×10^{-4}	2700	1.4	79	4.0×10^{-2}	130	6.5×10^{-2}
Regionalization or Centralization at Hanford	1.8	9.0×10^{-4}	2700	1.4	320	1.6×10^{-1}	560	2.8×10^{-1}
Regionalization or Centralization at Savannah River	1.8	9.0×10^{-4}	2700	1.4	320	1.6×10^{-1}	1700	8.5×10^{-1}
Regionalization or Centralization at Oak Ridge	1.8	9.0×10^{-4}	2700	1.4	320	1.6×10^{-1}	1700	8.5×10^{-1}
Regionalization or Centralization at Nevada Test Site	1.8	9.0×10^{-4}	2700	1.4	320	1.6×10^{-1}	1700	8.5×10^{-1}

90

transition shipments were included. Therefore, incorporating the transition period would actually reduce the difference between alternatives from the standpoint of transportation effects.

A.9 EFFECT ON ENVIRONMENTAL JUSTICE

The only method used to ship naval spent nuclear fuel to INEL in the past and the only method proposed for future shipments is by rail. The only exceptions to this are that naval spent nuclear fuel from Pearl Harbor Naval Shipyard is transported by ship from Hawaii to Puget Sound Naval Shipyard where the shipping containers are transferred to railcars for the journey to INEL, and a heavy-lift transporter is used to move the shipping containers from the Kesselring Site a few miles to the nearest railhead. The mode of shipment used for naval spent nuclear fuel tends to limit the exposure to members of the general public during transportation. The shipments pass through urban, suburban, and rural areas, using routes selected by the railroads in accordance with applicable regulations and the requirements of the load. The fractions of the distance traveled in urban, suburban, and rural areas range from about 2.5% urban, 12.5% suburban, and 85% rural to approximately 4% urban, 35% suburban, and 61% rural, depending on the alternative considered.

As shown in the analyses in this Attachment, the impacts on human health or the environment resulting from routine transport of naval spent nuclear fuel and hypothetical transportation accidents would be small for all of the alternatives considered. For example, it is unlikely that a single additional cancer would occur as a result of the transportation of naval spent nuclear fuel under any alternative. Shipping accidents could occur at any location along the routes used, so it is not possible to identify the minority or low-income composition of the populations along the routes. However, the fact that the potential impacts due to an accident for any of the alternatives considered would present no significant risk and do not constitute a credible adverse impact on the population along the shipping routes makes it possible to state that no adverse effects from accidents associated with the management of naval spent nuclear fuel would be expected for any specific segment of the population, minorities and low-income groups included.

To place the impacts on environmental justice in perspective, the risk from routine shipping activities or hypothetical accidents associated with transportation of naval spent nuclear fuel under any of the alternatives considered would amount to less than one additional fatality per year in the entire population. For comparison, in 1990 there were approximately 40,000 traffic fatalities in the United

States population and there were about 7,400 deaths caused by traffic accidents among people of color in the U. S. Even if all of the additional cancer deaths associated with an accident for any of the alternatives considered for naval spent nuclear fuel management were assumed to occur only among people of color, that group would experience far less than one additional fatality per year. The same conclusion can be drawn for low-income groups.

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ATTACHMENT B - DESCRIPTION OF NAVAL SPENT NUCLEAR FUEL RECEIPT AND HANDLING AT THE EXPENDED CORE FACILITY AT THE IDAHO NATIONAL ENGINEERING LABORATORY

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont)

LIST OF FIGURES

ATTACHMENT B

DESCRIPTION OF NAVAL SPENT NUCLEAR FUEL RECEIPT AND HANDLING AT THE EXPENDED CORE FACILITY AT THE IDAHO NATIONAL ENGINEERING LABORATORY

B.1 GENERAL DESCRIPTION AND OPERATION OF FACILITIES

The Expended Core Facility (ECF) is located within the confines of the Naval Reactors Facility (NRF) at the Idaho National Engineering Laboratory (INEL). It is a large laboratory facility used to receive, examine, prepare for storage, and ship naval spent nuclear fuel and irradiated test specimen assemblies. The information derived from the examinations performed at ECF provides engineering data on nuclear reactor environments, material behavior, and design performance. These data are used to develop new technology and to improve the cost-effectiveness of existing designs. Naval spent nuclear fuel is prepared at ECF for storage and shipment to the Idaho Chemical Processing Plant (ICPP). Some naval equipment contaminated by radioactive material during use in the fleet is refurbished for reuse.

The building which houses ECF is a concrete block structure approximately 1000 feet by 194 feet. This space provides offices and enclosed work areas, including an array of interconnected reinforced concrete water pools which permit visual observation of naval spent nuclear fuel during handling and inspection while shielding workers from radiation. Adjacent to the water pools are shielded cells used for operations which must be performed dry. Access to ECF for receipt and shipping of large containers is provided by large roll-up doors that allow railcar and truck entry. A schematic view of ECF is shown in Figure B-1 and a photograph of the water pool area is provided in Figure B-2.

ECF has been specifically designed to provide the unique physical and administrative controls required by the Naval Nuclear Propulsion Program to ensure safe handling of irradiated and contaminated nuclear fuels and components with a high degree of worker safety and protection for the

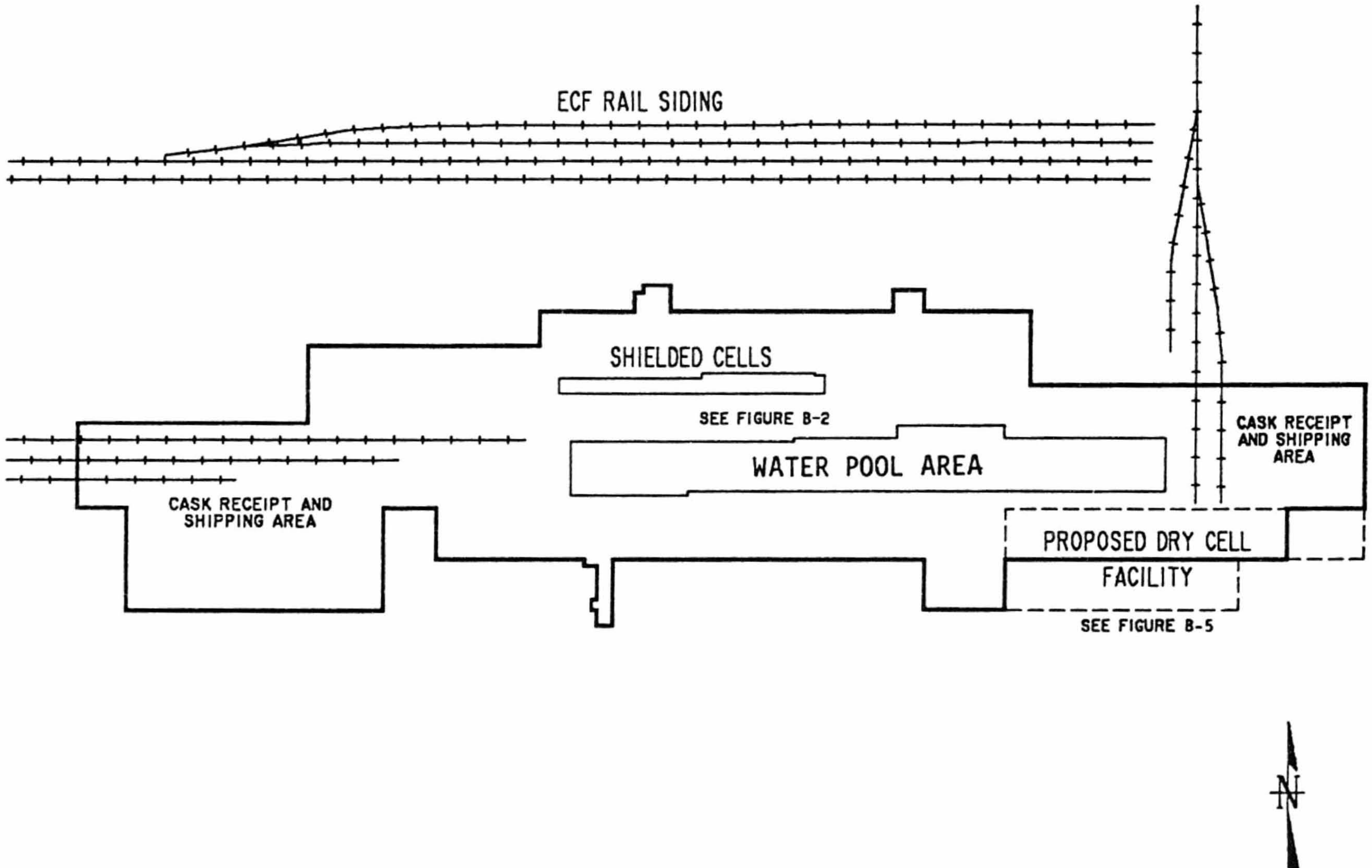

Figure B-1. Schematic view of Expended Core Facility.

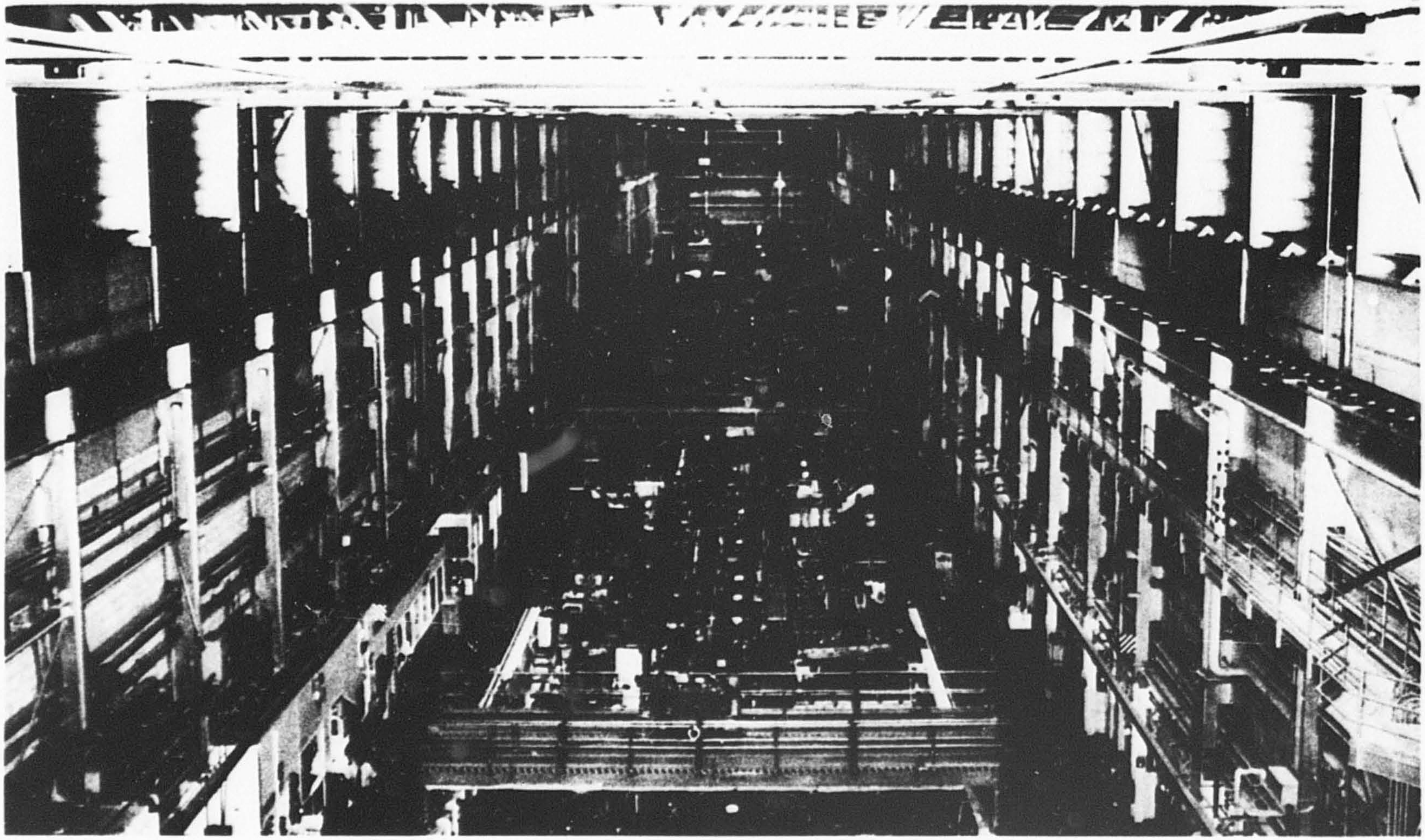

Figure B-2. Expended Core Facility water pool area.

98

environment. The original ECF building was constructed in 1957, and consisted of a water pool and a shielded cell with a connecting transfer canal. The facility has been modified as necessary to accomplish the expanding mission of the facility since then, including the addition of three more water pools, several shielded cells, and other capabilities dictated by the nature of the work required.

B.1.1 Water Pools

The purpose of the four interconnected water pools is to permit viewing and examination of radioactive reactor components and specimens while providing radiation shielding for workers.

Walls and stainless steel gates divide the water pools into smaller work areas called zones. This partitioning makes it possible to drain a small portion of the total water pool volume when facility equipment maintenance or repair is required. It also would permit isolation of an individual zone if a leak were to develop which, combined with transfer of the water from that pool to holding facilities, would minimize the loss of water.

B.1.1.1 Water Pit No. 1. This pool is used for the removal of spent fuel from shipping containers, and for preparation of fuel and low-level waste for shipment to ICPP. It also contains fuel and non-fuel storage areas.

B.1.1.2 Water Pit No. 2. This water pool is used for handling irradiation test assemblies. Various components are tested for their reaction to radiation. Test assemblies returned from the Advanced Test Reactor (ATR) at INEL are unloaded from the shipping cask and disassembled. Verification of test integrity and connection of electrical and mechanical monitoring devices are performed.

B.1.1.3 Water Pit No. 3. Radioactive components are separated by milling machines into smaller units for examination in this water pool. Dimensional measuring equipment is used to examine selected components. Fuel storage racks are also located in Water Pit No. 3.

Observation rooms are located along the northern wall of this water pool. These rooms are below the level of the water surface and have viewing windows into the water pool. Components may be visually examined and remotely handled underwater for shielding purposes from these rooms.

B.1.1.4 Water Pit No. 4. Operations performed in this water pool include spent fuel removal from transfer containers, temporary fuel storage in racks, fuel examination, and preparations for spent fuel shipments. Observation rooms are located along the northern wall of the water pool. This water pool also contains the transfer canals that would link the water pools with the proposed Dry Cell Project, which would prepare spent fuel for shipment in a dry, enclosed environment.

B.1.1.5 Construction. All of the water pools are constructed of reinforced concrete in such a manner that they are watertight. The water pool floors are designed to support installed equipment and shielded shipping containers weighing up to 100 tons with a minimum base area of 8 square feet. Water pool zone depths range from 20 feet to 45 feet. Water pool walls and floors are coated with a thermo-setting plastic coating which is highly resistant to radiation damage, is easy to decontaminate, and serves as an extra barrier to water leakage.

B.1.1.6 Water Treatment and Minimizing Radioactive Contamination. Radioactive contaminants which have accumulated in the ECF water pools through the introduction of corrosion products from irradiation test assemblies and the unloading of spent fuel are removed by various filtration techniques. The design basis for the ECF water treatment system is to allow no discharge of radioactive material to the environment, maintain water clarity, and minimize the amount of radioactive contaminants in the water.

The design goals are accomplished through the use of water purification modules, water pool surface skimming to remove film and floating material, and water recycling systems. The water purification modules prefilter the water to remove particles larger than 60 microns in diameter, remove any dissolved solids in ion-exchange resin beds, and remove any organic or suspended material by absorption in an activated carbon bed. Spent resin, carbon, and filter elements are disposed of as solid radioactive waste.

B.1.1.7 Water Management. The total volume of the ECF water pools (excluding the two new transfer canals that are empty) is 3,000,000 gallons. A 1-inch difference in the water pool level is equivalent to approximately 9,300 gallons.

The water pools are maintained at a nearly constant level. Alarms are installed to indicate both high and low level conditions. The total water volume is accounted for monthly. Any addition of water to the system is reported to a separate NRF site organization for an independent verification of water volume.

Water leaves the water pools via evaporation, temporary filling of shipping containers, decontamination of equipment, and transfers to retention basins. The water pool evaporation rate has been calculated theoretically and confirmed by experiment. Water returns to the water pools by transfers from the retention basins and by draining shipping containers. Water removed from the system due to evaporation and equipment decontamination is replaced by adding demineralized water.

ECF has the capability of storing 235,000 gallons of water pool water in three underground, steel-reinforced, concrete storage basins. Two of the vaults each have a 40,000-gallon capacity, and the third has a 155,000-gallon capacity. These basins provide the capability to replenish the water pools and receive water pool water if draining a water pool zone is necessary.

B.1.2 Shielded Cells

There are 14 concrete shielded cells in the facility. These shielded cells are used for examination of smaller components, such as specimens which have been removed from irradiation tests that have been exposed to a neutron flux in the ATR, and fuel and non-fuel components from the water pools.

The shielded cells are constructed of concrete, with walls 3 feet thick to provide shielding from radiation. Ventilation in the cell bank maintains negative pressure inside the cells in relation to the rest of the facility. This ensures that radiological contamination is contained within the cells.

All work in the shielded cells is performed remotely by equipment controlled from the cell gallery, and is viewed through shielded lead glass windows. The windows are 3 feet thick, and provide the same shielding value as the concrete walls. The interior of the cells can also be viewed through wall periscopes that permit undistorted viewing of equipment and components.

B.2 RECEIPT AND HANDLING OF NAVAL SPENT NUCLEAR FUEL

B.2.1 Receipt of Spent Fuel

Nuclear-powered ship assignments for refueling, defueling, and overhaul are currently performed by the six nuclear-capable public shipyards (Mare Island, Puget Sound, Pearl Harbor, Portsmouth, Norfolk, and Charleston) and one nuclear-capable private shipyard (Newport News). In 1993, the federal base closing commission included Mare Island and Charleston Naval Shipyards among the bases to be closed in the near future. The spent fuel is removed from nuclear-powered ships and loaded into shipping containers designed specifically for naval spent nuclear fuel. The spent fuel containers are loaded and sealed at the shipyard and shipped to ECF via railcars, as described in Attachment A. A maximum of 48 containers can be staged on the rail siding at NRF outside ECF while awaiting transfer of the spent fuel to the water pools. ECF also receives spent fuel from naval prototype plants in a similar manner.

B.2.2 Handling of Spent Fuel

The shipping containers are brought into the ECF building at one of the two defueling stations and are prepared for defueling by removing the dust cover, leveling, and filling with water. Appropriate containments to prevent release of radioactive material are installed and the container access plug is removed to allow access to the fuel modules.

The containers are unloaded at either the west end defueling station or the east end defueling station. Regardless of the defueling station used, the fuel modules are removed from their shipping container one at a time using a fuel handling machine which draws the module out of the container into a shielded volume, and the entire machine is transferred to the water pools. The fuel module is then discharged into a receiving receptacle in the water pools. Photographs of the two fuel handling machines used are provided in Figures B-3 and B-4.

Every item containing nuclear fuel received at ECF has a unique serial number. When the fuel is removed from its shipping container, two ECF fuel handlers independently read the serial number and compare it to the shipping paperwork. After the serial number is confirmed, the fuel is

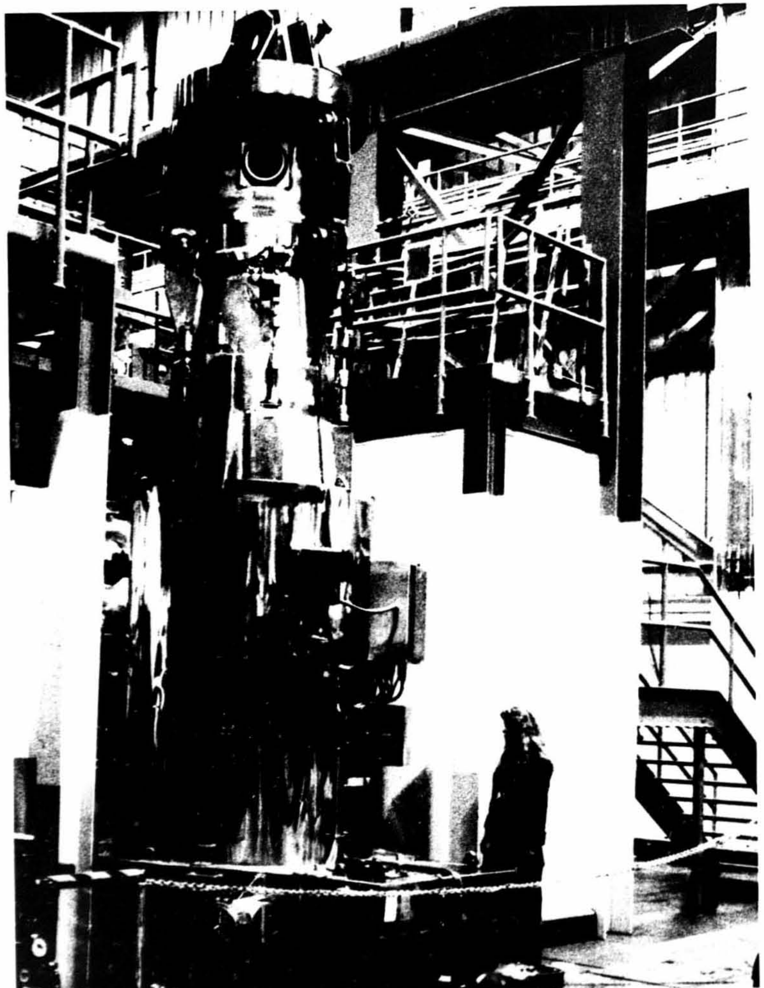

Figure B-3. M-140 container fuel handling machine.

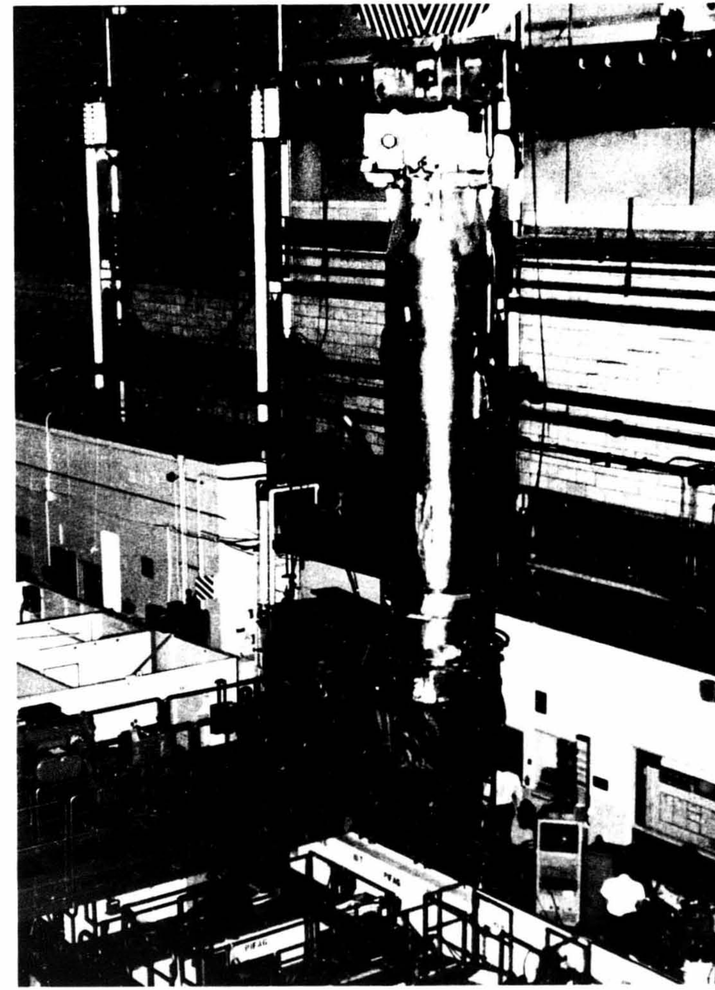

Figure B-4. M-130 container fuel handling machine.

moved to a uniquely numbered storage port location. Two fuel handlers then independently verify that the fuel is stored in the correct storage location. ECF has a computer-based fuel accountability system which maintains a record of the location and type of every piece of nuclear fuel and how many grams of uranium are contained within the fuel. This system tracks every fuel movement during the time that the fuel is at ECF.

All naval fuel modules have metal structures which contain no fuel above and below the fuel region to facilitate coolant flow and maintain proper support and spacing within the reactor. These upper and lower non-fuel bearing structures must be removed to provide access to the fuel-bearing sections to permit inspection of the module. Removal also reduces the storage space ultimately required for the fuel by approximately 50 percent. The upper and lower non-fuel bearing structures removed during the preparation of fuel modules are evaluated using the waste classification criteria established by federal regulations in 10CFR61 and DOE Order 5820.2. These non-fuel bearing structures do not contain any fuel, or fission products from fuel, and therefore cannot be considered "spent nuclear fuel." They also do not contain transuranic elements or fission products and thus cannot be considered high-level waste or transuranic waste. Therefore, the amounts of radioactivity in the end boxes cause them to be classified as low-level waste. As indicated in Section 5.2.15, the amount of low-level waste generated each year at the Expended Core Facility is 425 cubic meters. The radioactive isotopes which represent 99 percent of the activity in this material are identified as follows:

ISOTOPE	HALF-LIFE (Years)	PRIMARY MODE OF DECAY
Fe-55	2.73	Electron Capture (x-ray)
Co-60	5.271	Beta and Gamma
Ni-59	76,000	Electron Capture
Ni-63	100	Beta

U.S. Nuclear Regulatory Commission 10CFR61 identifies three classes of low-level wastes which are generally suitable for near-surface disposal, namely, Classes A, B, and C. Those meeting the requirements for near-surface disposal are shipped to the INEL Radioactive Waste Management Complex using a shielded cask. Wastes with concentrations greater than those specified for Class C for certain short- and long-lived isotopes were found to be not generally suitable for near-surface disposal. These wastes are classified as Greater Than Class C Low-Level Radioactive Waste. In May 1989, the Nuclear Regulatory Commission promulgated a rule that requires disposal of

commercially generated low-level waste with concentrations of radioactivity greater than Class C in a deep geologic repository, unless disposal elsewhere is approved by the Nuclear Regulatory Commission.

Currently, a small amount (about 25 cubic meters) of greater than Class C low-level waste in material removed from the ends of naval spent nuclear fuel modules over the years is being stored at the Naval Reactors Facility pending availability of a disposal facility licensed by the Nuclear Regulatory Commission. This material has been collected and held at the Expended Core Facility for many years. This practice is expected to continue over the period of time covered by this Environmental Impact Statement.

After these upper and lower metal structures have been removed from a fuel module, a lifting fixture is installed to facilitate handling. Prepared fuel may then be inspected immediately or it may be held for a time prior to inspection in storage racks in the water pool. In the event that the fuel is temporarily stored while awaiting inspection, spacers are placed at the bottom of the selected port in the storage rack to maintain the position of the fuel module close to the top of the rack to make movement of the module easier.

Visual examinations of all modules are performed to verify that the fuel has performed as expected. As discussed in Section 2.4.1, about 10 to 20 percent of the spent reactor cores are selected for more detailed examination or destructive analysis in accordance with the needs of the Naval Reactors fuel development program. The more extensive examinations performed in the water pools include measurements of key dimensions of the modules and collection of specimens to be examined in the shielded cells. The specialized equipment used to perform examinations of naval spent nuclear fuel are described in more detail in the section of this attachment devoted to equipment. Destructive analyses are performed at the Expended Core Facility or at other laboratories, but all material subjected to such analysis must be removed from the spent fuel modules at the Expended Core Facility.

The last steps of spent fuel handling performed at ECF are staging the module for shipment and loading the module into the shipping cask used to transport spent fuel from ECF to ICPP. The spent fuel may be temporarily stored in the racks in the ECF water pools until a cask becomes available to transfer the material to ICPP.

B.2.3 Shipment of Fuel to the Idaho Chemical Processing Plant

A lead-filled, stainless steel shipping cask is used to transport naval and prototype spent fuel modules from ECF to ICPP. The cask is removed from its transport truck and lowered into the ECF water pool until it rests on the floor of the pool. The closure head is removed, and inserts are placed in the cask to provide proper spacing of fuel and to maintain proper positioning during transport of the modules. The modules are inserted into the cask, the closure head is reinstalled, and the cask is lifted from the water. The cask is drained, the exterior is decontaminated, and the cask is loaded onto the truck for shipment. The transport of the cask to ICPP is described in Attachment A.

B.2.4 Library of Naval Reactor Components

As the first modules of a given fuel design are received at the Expended Core Facility for examination, selected key operating components are retained in "library" storage in the water pools to provide a source of reference. These older components are kept to ensure that there will be a representative item available to assist in diagnosis of problems which may occur in any operating power plant in the fleet. The items chosen for this library are usually those that have been in service the longest so that they display the most pronounced effects of use. As the various fuel design types are replaced in fleet service by newer designs, fuel components related to the fuel design being retired are removed from library storage and shipped to ICPP.

B.3 HANDLING OF IRRADIATED TEST SPECIMENS

The irradiated materials program evaluates small specimens of materials for use in naval reactor systems. The specimens are loaded in sample holders, and the holders are placed in test assemblies at ECF. The assemblies are irradiated at ATR, and returned to ECF for disassembly. The specimens are cleaned, examined, reloaded in a test assembly, and returned to the ATR for continued irradiation. A typical specimen undergoes several cycles of irradiation and examination over several months or years. Examinations include nondestructive and destructive tests. Destructive tests have historically included sectioning of specimens for mechanical testing and metallography. Metallographic work was performed in the ECF hot cells in the past and is planned to be performed on specimens in the future.

After completion of the final examination, specimens are shipped to ICPP for storage or to the INEL Radioactive Waste Management Complex for disposal. Other specimens are shipped to either the Bettis Atomic Power Laboratory near Pittsburgh, Pennsylvania, or the Knolls Atomic Power Laboratory near Schenectady, New York for more detailed examinations.

B.4 DESCRIPTION OF MAJOR ITEMS OF EQUIPMENT

The normal method for moving the fuel in the water pools to designated examination equipment areas is by use of one of five bridge cranes which move on rails located on the tops of the walls of the water pools. The fuel is handled remotely. All fuel movements are controlled by trained personnel, and accountability is maintained both by computer and by personnel using fuel transfer forms.

B.4.1 Water Pool Equipment

ECF has unique equipment in the water pools that has been designed for remote operation underwater to perform specific examinations on naval spent nuclear fuel and irradiated test specimens. Special consideration was given during equipment design to provide for remote repair and replacement of components. A description of the water pool spent nuclear fuel and irradiation test examination equipment is presented below.

B.4.1.1 Water Pool Band Saws. There are two underwater band saws in the ECF water pools. These band saws are used to remove the non-fuel bearing structural material from the top and bottom of fuel cells in preparation for inspection. The fuel region of the fuel cell remains intact during the cutting procedure.

B.4.1.2 Water Pool Milling Machines. Three milling machines in the water pools are used to separate spent nuclear fuel components into smaller sections for examination in the shielded cells. The fuel region of the fuel cell remains intact during the machining. The mills are used to section spent fuel into pieces which can be handled in the shielded cells for examinations, such as gamma radiation measurement, or for obtaining smaller specimens for metallurgical analysis or fuel depletion measurement. The mill head of the largest milling machine can be remotely interchanged with a band saw attachment to convert the machine into a cutoff saw.

B.4.1.3 Universal Inspection Station. This equipment is used to obtain dimensional measurements using specially designed probes that are inserted in the fuel module. This equipment can position and rotate the probe in any orientation by a dedicated computer. This information is used to assess dimensional changes in the fuel module.

B.4.1.4 Vertical Inspection Gage. The vertical inspection gage is used for obtaining dimensional measurements or to trace the contour of the external surfaces of fuel cell assemblies or control rods. This information can be used to provide a three-dimensional image of the fuel cell or control rod at the end of fuel life to determine the effects of fuel element changes on the overall fuel cell assembly dimensions over fuel life and the effects of radiation on control rod dimensions over fuel life.

B.4.1.5 Video Visual Equipment. Underwater television cameras and lighting can be set up in any zone in the water pools to obtain images of the external surfaces of the fuel cell assemblies and control rods. These visual inspections are used to search for anomalies such as excessive corrosion or wear on external surfaces. The bottom end of the fuel cell assemblies can also be inspected for flow blockage, corrosion, and wear.

B.4.1.6 Assembly and Disassembly Tables. These tables are used to assemble and disassemble irradiated test assemblies that are inserted in the ATR. There are two identical assembly and disassembly tables installed side by side in the water pools. Each is mounted on a tilt platform that is used to rotate the table from a horizontal position for test assembly and disassembly to a vertical position for loading and unloading the test assembly.

B.4.1.7 Headwork Station. The Headwork Station provides containment and shielding for the mechanical connection and disconnection of components to and from the unirradiated portion of the assembly and disassembly of irradiations tests for the ATR. There are two independent work stations; each consists of an elevator platform which raises the top unirradiated portion of the test above the water surface. A containment is positioned above the water surface to prevent the spread of contamination while the examination is performed above the water.

B.4.1.8 Fuel Storage Racks. Storage racks are required at ECF since, at times, fuel is received into the facility faster than fuel can be prepared and shipped out of the facility. Racks are also used to store the small amount of naval spent nuclear fuel selected for retention as library specimens for

future reference and study. Ensuring that the racks are conservatively designed to withstand any credible accident and continue to provide adequate nuclear separation are the major criteria for storage racks.

The basic configuration of a fuel storage rack is a rectangular structural array of storage ports. Each port has a square opening, but depth is variable. All storage ports in use at ECF are stainless steel. Stainless steel is used exclusively to resist corrosion during the life of the storage racks. The storage ports are designed to withstand the weight of the heaviest fuel module which can be placed in the port, and the frame assembly is designed to support the entire weight of all the fuel ports fully loaded with the heaviest fuel type.

All the fuel racks are designed to maintain their structural integrity during a design basis earthquake and to withstand the impact of a fuel module dropped onto the fuel racks. Analyses of all fuel racks in the event of seismic activity has demonstrated that they will not collapse during the postulated earthquake. ECF also performed a full analysis of the strength of the ports if a fuel module were dropped over the fuel racks, including the kinetic energy which the dropping fuel module would impart to the rack. It was determined that all fuel racks at ECF were adequately designed to withstand the energy of dropped fuel. The analysis also identified that some equipment handled at ECF was heavy enough that the racks might be deformed if the equipment were dropped. Thus, operating rules and procedures prohibit the movement of large loads over the fuel racks to ensure that no accidental damage to the racks can occur.

Fuel storage racks were also designed to prevent arrangement of the modules into a potentially critical configuration. The fuel racks are designed so that each port separates the module it contains from every other module by a distance great enough to prevent criticality under the most limiting conditions possible. To assure that only one piece of fuel is placed in a port, all fuel storage ports are equipped with lids which can be locked and sealed. Finally, the frame assemblies of all fuel storage racks are covered with stainless steel sheeting to prevent fuel from inadvertently being placed between fuel storage ports.

B.4.2 Water Pool to Shielded Cell Transfer Systems

Components that have been removed from spent nuclear fuel cells or test assemblies can be transferred into the shielded cells using one of the three available water pool to shielded cell transfer systems. The transfer systems use carts that are driven through underwater tunnels.

B.4.3 Shielded Cell Examination Equipment

ECF has specialized equipment installed in the shielded cells which is designed to perform examinations on fuel elements and components removed from spent fuel cell assemblies and test specimens that have been irradiated in the ATR. A description of the major shielded cell equipment follows.

B.4.3.1 Electronic Balances. These are commercially available electronic balances that have been modified to operate remotely in the shielded cells. Components on these balances that are known to deteriorate from exposure to radiation have been replaced using materials that are less susceptible to radiation damage. The equipment is interfaced with computer data acquisition systems to aid the operators in tracking and reducing the data. These balances are used primarily to assess weight changes that result from corrosion testing of materials in the ATR.

B.4.3.2 Descale Tanks. Corrosion removal is performed for test specimens that have been irradiated in the ATR and structural components and fuel elements removed from spent nuclear fuel modules. These tanks use heat, chemicals, and ultrasound to dislodge corrosion that has accumulated on the specimens or components. The corrosion removal aids in visual examination of these specimens.

B.4.3.3 Bridgeport Milling Machine. This is a high-precision milling machine that has been modified for remote operation in the ECF shielded cells. The mill is controlled by a programmable controller located in the shielded cell gallery. The Bridgeport mill is used for precise machining of non-fuel components removed from spent nuclear fuel cell assemblies.

B.4.3.4 Specimen Coordinate Automated Measuring Machine. The specimen coordinate automated measuring machine is a fully automated unit specifically designed to perform three-

dimensional measurements on irradiated test specimens and structural components removed from spent nuclear fuel cells. The equipment is completely computer controlled and has an accuracy of 0.00005 inch (50 microinches). The information obtained from this equipment is used to assess the effect of radiation on material growth and fuel burnup on swelling of specimens.

B.4.3.5 Fiducial Automated Measuring Machine. This machine is used to measure the distance between scribe marks that are put on some types of specimens during fabrication. The machine accurately measures the position of the scribe marks in relation to other fiducial marks on the specimen. These data are used to assess the effects of radiation on specimen growth and distortion, as well as the effect of fuel depletion on fuel element swelling.

B.4.3.6 Gamma Scan System. This system measures gamma radiation emitted by fission products to identify isotopes present in the fuel as a result of fuel depletion. The system is controlled by a dedicated computer which positions the specimen, provides for data acquisition and evaluation, and provides an output of the isotopes detected by the system at each location along the axes of the specimen.

B.4.3.7 Alpha Box. The Alpha Box is a carbon steel containment inside the shielded cells. It provides isolation within the shielded cells for fuel cutting to prevent the spread of fission products. This is the only location in the facility where cutting through the fuel region of spent nuclear fuel is allowed.

B.5 FACILITY DESIGN AND INTEGRITY REQUIREMENTS

B.5.1 Flood

A flood at ECF due to overflow of any source of surface water within the INEL boundaries is a low probability event. With the construction of the INEL flood control diversion system in 1958, the threat of a flood from overflowing of the Big Lost River, the primary source of surface water at the INEL, has become very small.

The maximum water elevation postulated at ECF would be caused by a hypothetical Probable Maximum Flood resulting from failure of the Mackay Dam, located approximately 35 miles northwest of the INEL. The hypothetical flood could result in a maximum water level approximately 3 feet above the floor elevation of the ECF building. This flood is postulated to result from water flowing over the top of the Mackay Dam and causing it to fail due to high water levels. This flood is highly unlikely. (Koslow and Van Haaften 1986)

Dam failure due to other causes, such as seismic activity, is more likely. Although the Mackay Dam survived the 1983 Borah Peak earthquake without damage, it was built without seismic design criteria. Additionally, it is not clear how resistant the dam structure is to seismic events. A fault segment runs within 6 kilometers of the Mackay Dam.

Flooding of the ECF building is possible should the Mackay Dam fail. Flooding of the ECF building would not create a nuclear criticality hazard. Flooding of the building could result in the release of water containing low levels of radioactive contamination to the environment and damage to equipment in flooded areas. Following the dam break, it would take over 16 hours for the flood water to reach NRF. This is adequate time to complete emergency procedure preparations, such as filling and placing sandbags, for the expected flood conditions.

B.5.2 Earthquake

The ECF building structure was built in accordance with the Uniform Building Code for each particular phase of construction. Water Pit No. 1, Water Pit No. 2, and Water Pit No. 3 were built to "Zone 2" earthquake requirements which were judged to be appropriate under the U.S. Geologic Survey classification of the area at the time of their construction. Water Pit No. 4 and its two transfer canals were built to the more restrictive "Zone 3" earthquake requirements in effect at the time they were built.

A seismic assessment has been performed for the ECF using the actual characteristics of the existing facility. Based on this assessment, a design basis seismic event at ECF could have a peak ground acceleration of 0.24 g (Rizzo 1994). This peak ground acceleration is derived on the basis that a moment magnitude 6.9 seismic event centered near Howe on the Lemhi fault would cause a rupture of approximately 34 kilometers along the Lemhi fault. The Howe epicenter is the epicenter located closest to ECF, and 6.9 was the moment magnitude of the Borah Peak earthquake in 1983. This approach for postulating the location of the seismic event is consistent with the Nuclear

Regulatory Commission methodology used for commercial power plants. The beyond design basis seismic event was based on a scenario resulting in a peak ground acceleration of 0.4 g at ECF.

B.5.3 Tornado

A tornado at ECF is a low probability event. The document "Technical Basis for Interim Regional Tornado Criteria," WASH-1300, provides the technical basis for Nuclear Reactor Commission Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The WASH-1300 document identifies the probability of occurrence of a tornado at ECF to be 7.8×10^{-5} per year based on historical records. Regulatory Guide 1.76 identifies the maximum wind speed appropriate to ECF to be 240 mph. Data collected by Dr. T. Fugita of the University of Chicago performed at the request of the DOE for the period between 1950 and 1976 indicate the probability of a tornado with winds of that speed occurring at the INEL is about 1.3×10^{-9} per year. Based on a threshold wind speed for tornado damage of 75 mph (refer to P. L. Doan, "Tornado Considerations for Nuclear Power Plant Structures," Nuclear Safety, Volume 11, No. 4) and a probability of 0.80 for the occurrence of tornado-induced wind speeds greater than or equal to 75 mph (WASH-1300, Table 3), the probability of a damaging tornado occurring at ECF is 7.8×10^{-5} per year x 0.80 = 6.2×10^{-5} per year.

A tornado could not affect the fuel storage area in ECF in such a way that the fuel would be rearranged into a critical configuration. The article by Doan cited above analyzes the effects of tornados for the general case of spent fuel in water pools and concludes "... massive loss of water due to either tornado-induced wind forces or tornado-generated missiles cannot happen. It is credible, however, that a couple of feet of water could be lost owing to the combination of water splashing, water entrainment, and pressure differentials. The spent fuel at the bottom of the water pools would, however, remain completely covered.... By the same token, the radiation dose level above the water surface would not increase by any meaningful amount."

B.5.4 Fires

The entire ECF facility is protected against fires by one of several types of sprinkler systems. A large, intense fire in fuel handling areas is a low probability event because of the nature of the materials of construction in these areas, the amounts and kinds of material present, and the fire protection system. Most of the spent fuel is under many feet of water, providing additional protection against a fire which might involve fuel. Fires at other locations in the facility would be extinguished by the sprinkler system and by manual fire protection equipment (e.g., fire extinguishers or fire hoses). An extensive fire involving the ECF building structure is highly unlikely because it has been constructed of non-combustible or fire-resistive material to the greatest extent possible, in accordance with applicable Atomic Energy Commission, Energy Resource and Development Administration, and DOE design criteria.

B.5.5 Loss of Water Pool Water

Loss of all water in a section of the water pool is extremely unlikely. However, should a heavy object be dropped onto a water pool floor, a crack could develop. If this were to occur, the cracked water pool area would be isolated and drained in a controlled manner to one of the retention basins before a substantial loss of water to the environment would occur. Even in the event that severe damage to a water pool floor were to result in the loss of substantial amounts of water pool water, no nuclear criticality hazard would result and no melting of fuel would occur.

B.6 CRITICALITY CONTROL

There has never been an inadvertent criticality at the Expended Core Facility. This is the result of strict application of the following principles.

A fundamental principle of nuclear safety is Criticality Control. When a mass of nuclear fuel reaches a condition at which its atoms are capable of undergoing a self-sustaining chain reaction, or splitting (fissioning) into new elements, the result is called a criticality. Nuclear fission releases energy in the form of radiation and heat. Controlled criticality within a shielded reactor vessel produces energy within a confined space without harm to personnel or the environment. Although the water pools, the shielded cells, and the ECF building are designed to shield and contain radiation and radioactive contamination, an uncontrolled criticality (or nuclear excursion) within ECF is unacceptable, and comprehensive measures are taken to prevent such an occurrence. Criticality control at ECF could be described more accurately as "absolute criticality prevention." Conditions are

identified, equipment or processes are designed, rules and procedures are formulated, and personnel are trained to prevent occurrence of an accidental criticality.

Safety analyses are performed on all fuel types and system designs where all single plausible and unlikely accidents are considered. Conservatism is employed in establishing limits and controls, and spent fuel is handled to the more restrictive as-built values. Then a "double accident criterion" is applied to all fuel handling equipment and procedures. The double accident criterion states "Fuel must be handled and equipment designed so that acceptable margins to criticality exist after two most limiting, unlikely, independent, and concurrent accidents. In this context, two errors in a routine administrative procedure are considered to be a single accident, not two." As a result of application of this criterion to equipment and procedures at ECF, the amount of fuel which may be handled in any operation is typically restricted to one quarter of the minimum amount which could achieve criticality minus a safe margin to criticality.

All nuclear fuel operations must be performed in accordance with approved criticality control procedures. Nuclear safety analyses are carefully reviewed by the responsible management and two independent nuclear safety committees. Naval Reactors must approve each analysis before it is used. Strict reviews and approvals are also applied to implementation of safety analyses in fuel handling procedures.

The successful criticality control program at ECF is also due to thorough training and supervision of fuel handling personnel. Employees are educated concerning the principles of criticality, associated hazards, and prevention. A system of checks to ensure that the rules and limits are strictly observed is employed. It includes detailed training documentation, qualification and testing standards, a self-assessment (audit) program, and an array of accountability and nuclear safety drills.

B.7 PROPOSED DRY CELL FACILITY

The Dry Cell Facility consists of a shielded, radiologically controlled area with remotely operated equipment. The facility is designed for a 40-year life, built of structural steel and concrete, and would be integral with the existing ECF building.

The major element of the Dry Cell Facility is a large reinforced concrete shielded cell with interior dimensions of 22 feet wide by 84 feet long by 21 feet high, containing all the equipment necessary to inspect and disassemble fuel modules. The facility will have the capability to prepare and load one fuel module per shift in a shipping cask. Based on a two shift per day operation (500 shifts per year), and a 25-percent maintenance downtime, the Dry Cell Facility yearly capacity is expected to be 375 modules. Shielded decontamination and repair cells will be attached to the main shielded cell to allow remote decontamination and repair of equipment used throughout ECF. Artist's views of the Dry Cell Facility and the associated Cask Loading System are shown in Figures B-5 and B-6.

The dry cell design incorporates 4-foot thick, radiation shielding walls constructed of high-density and normal-density concrete. The shielding is designed to limit radiation levels in normally occupied areas around the cell to 0.1 millirem per hour or less. At the INEL Site boundary, there would be no measurable elevation above the naturally occurring background radiation levels. The dry cell design meets the latest seismic requirements and includes negative pressure air ventilation for radiological contamination control. Shielded lead glass windows and viewing aids are provided as required at the workstations. Power, lighting, and a fire suppression system are also provided.

The Dry Cell Facility is also designed to facilitate decontamination and decommissioning of the facility at some future date. This is achieved by including cell liner contamination barriers, no fixed embedded piping, a minimum of cracks and crevices, smooth surfaces, and wall penetrations large enough to be radiologically surveyed to verify decontamination effectiveness.

B.8 REFERENCES

Koslow, K. N. and D. H. Van Haaften, 1986, *Flood Routing Analysis for a Failure of Mackay Dam*, EGG-EP-7184, EG&G Idaho, Inc., Idaho Falls, Idaho, June.

Rizzo (Paul C. Rizzo Associates), 1994, *Natural Phenomena Hazards - Expended Core Facility, Idaho National Engineering Laboratory*, Monroeville, Pennsylvania, June.

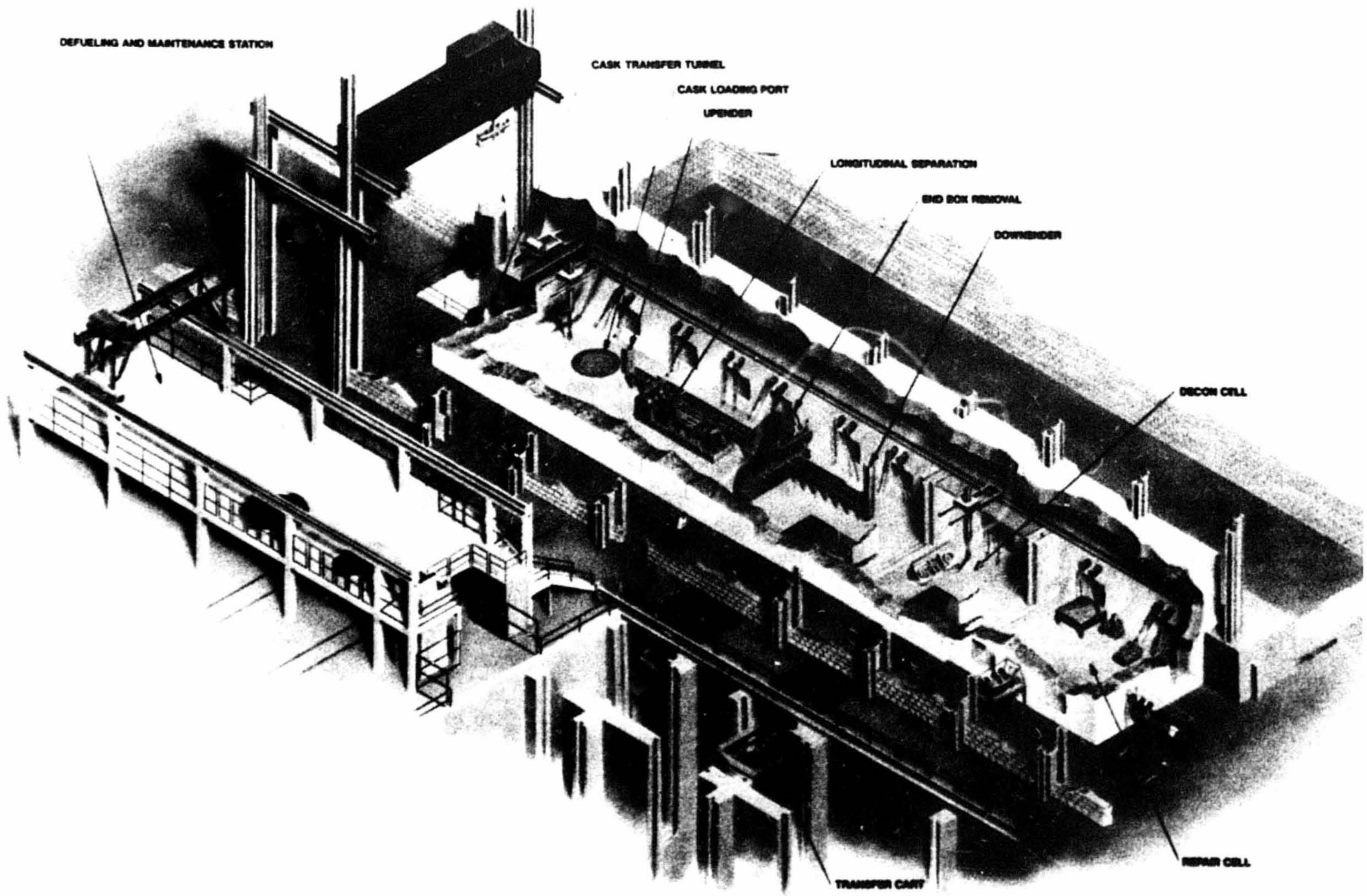

Figure B-5. Proposed ECF Dry Cell Facility.

118

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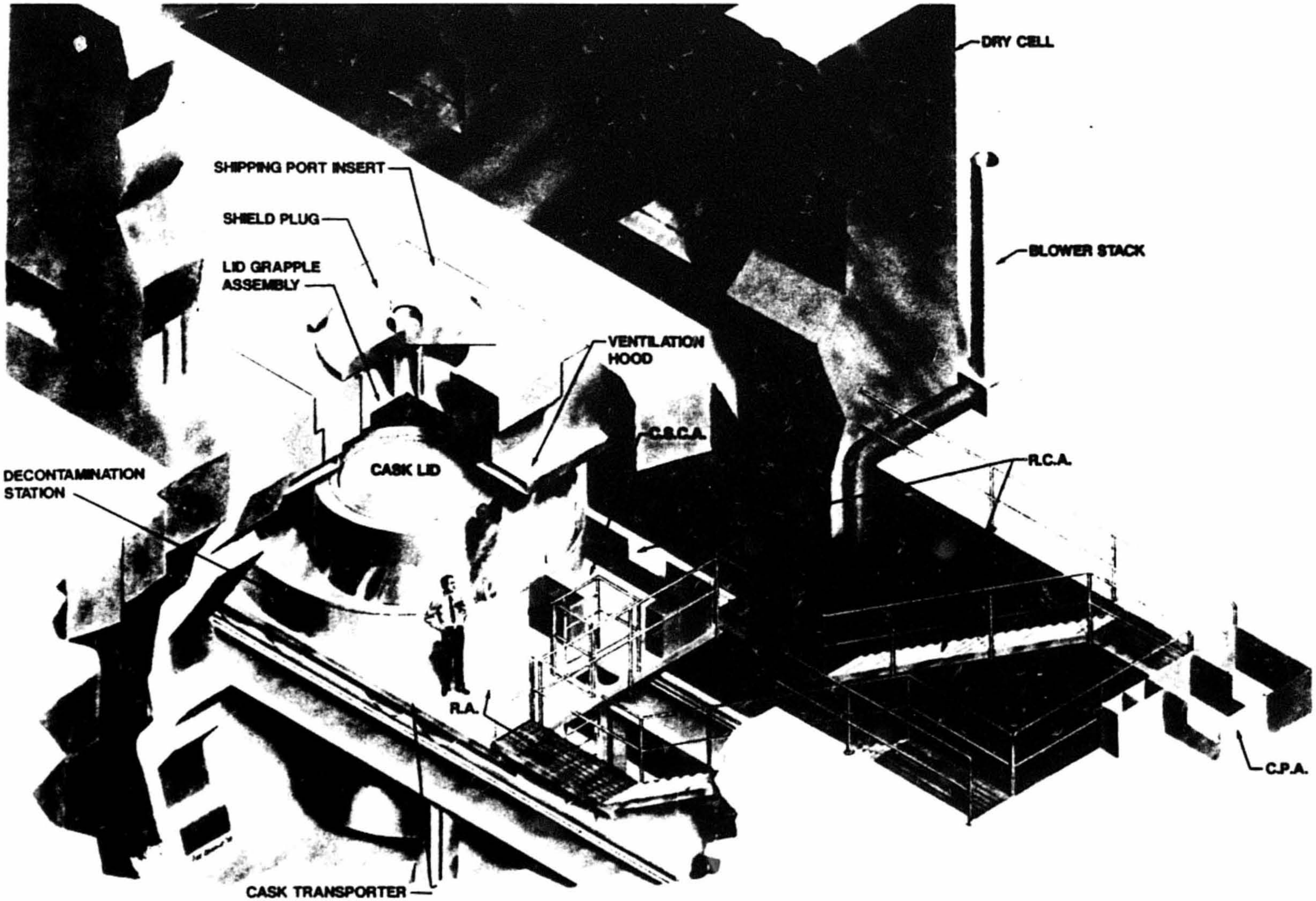

Figure B-6. ECF Dry Cell Facility Cask Loading System.

119

ATTACHMENT C - COMPARISON OF STORAGE IN NEW WATER POOLS VERSUS DRY CONTAINER STORAGE

TABLE OF CONTENTS

ATTACHMENT C

COMPARISON OF STORAGE IN NEW WATER POOLS VERSUS DRY CONTAINER STORAGE

C.1 INTRODUCTION

This attachment discusses the advantages and disadvantages of water pools versus dry container storage should construction of additional interim storage be required. The discussion considers the generic safety aspects of water pools and dry container storage based on evaluations performed by the Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) as well as experience with naval spent nuclear fuel.

C.2 WATER POOLS

During the last four decades, the Expended Core Facility (ECF) at the Idaho National Engineering Laboratory (INEL) has demonstrated the safety and reliability of water pools under the control of the Naval Reactors Program. Water pools have historically been the method of choice for interim storage and fuel handling because: (1) water has a high thermal capacity for the removal of heat from the fuel, (2) the transparency of water facilitates the inspection and movement of the fuel, (3) water is an excellent gamma and neutron shield, (4) water is easy to purify and recycle, and (5) water provides a means to prevent release of radioactive material into the air.

The safety of spent fuel storage in a water pool can be considered in terms of three generic criteria. They are: (1) the integrity of spent fuel under water pool storage conditions, (2) the structure and component safety of the facility, and (3) the potential risks of accidents and acts of sabotage at the spent fuel facility.

The NRC conducted an extensive investigation into the storage of spent fuel and documented the findings in the Waste Confidence Decision (NUREG 1984). Based on the technical evaluations cited in that document, the NRC found that the Zircaloy cladding which encases spent fuel is highly resistant to failure under pool storage conditions and concluded that Zircaloy-clad commercial fuel satisfied the first generic criterion. This conclusion is consistent with the extensive experience with naval spent nuclear fuel. Naval fuel is Zircaloy clad and thus is highly resistant to corrosion in water. In addition, a Navy fuel assembly has much higher mechanical integrity than commercial fuel since it is designed for military application and is capable of withstanding shock loadings which may be encountered in battle conditions.

The NRC also conducted an extensive evaluation of the structural and component safety of water pools. The NRC found no reason why spent fuel storage pools would not be capable of performing their cooling and storage functions for a number of years past the design life of 40 years if the water pools are properly maintained; therefore, the second generic criterion would be satisfied. This conclusion is consistent with the naval fuel experience of over 35 years of operation of the ECF.

The risk of major accidents at spent fuel storage pools resulting in off-site consequences is remote because of the secure and stable character of the spent fuel in the storage pool environment, and the absence of driving forces (i.e., high pressure or temperature) which might result in dispersal of radioactive material (NUREG 1984). The consequences of terrorist attacks on a spent fuel storage pool would be limited by the realities that the radioactive content of spent fuel is in the form of material encapsulated in high-integrity metal cladding and stored underwater in a reinforced concrete structure. Under these conditions, the radioactive content of spent fuel is relatively invulnerable to dispersal to the environment (NUREG 1984).

These considerations led the NRC to conclude that storage pools can be designed to safely withstand accidents caused either by natural or man-made phenomena such that there would be no impact to the environment. Therefore, the third generic criterion would be satisfied.

The NRC concluded that all areas of safety and environmental concern (e.g., maintenance of systems and components, prevention of material degradation, protection against accidents and sabotage) have been addressed for water pools, and that spent fuel can be stored with no environmental impact. This conclusion is supported by the Organization for Economic Co-Operation and Development of the Nuclear Energy Agency (NEA 1993).

C.3 DRY CONTAINER STORAGE

Dry container storage technologies have been in use in the United Kingdom since 1972 (MOCSG 1993). In the United States, demonstration projects have been underway since 1982. In dry container storage, multiple barriers prevent gaseous as well as particulate fission product releases. Two separate barriers must fail before fission products can be released: (1) the fuel cladding, and (2) the outer secondary seal. In addition, dry storage systems provide metal or concrete shielding to reduce the external radiation to acceptable limits.

The NRC concluded that dry container storage involves a simpler technology than that represented by water storage systems. Water storage relies to a certain extent upon active systems such as pumps, renewable filters, and cooling systems to maintain safe storage. Favorable water chemistry must also be maintained to retard corrosion. Dry container storage uses convective circulation of an inert atmosphere in a sealed dry system so there is little opportunity for corrosion (NUREG 1984).

The NRC also found that dry container storage of spent fuel in dry wells, vaults, silos, and metal casks is relatively invulnerable to sabotage and the forces of nature, because of the weight and size of the sealed, protective enclosures, which may include 100-ton steel casks, large concrete-lined casks, and surface concrete silos (NUREG 1980).

The NRC concluded that for dry interim storage, all areas of safety and environmental concern (e.g., maintenance of systems and components, prevention of material degradation, protection against accidents and sabotage) have been addressed and shown to present no more potential for adverse impact on the environment and the public health and safety than storage of spent fuel in water pools. This conclusion is supported by the Organization for Economic Co-Operation and Development of the Nuclear Energy Agency (NEA 1993).

As stated earlier, naval fuel uses Zircaloy cladding and has a much higher mechanical integrity than commercial fuel since naval fuel is designed for military application. Therefore, the generic conclusions reached for commercial spent fuel are directly applicable to naval spent fuel.

C.4 NON-RADIOLOGICAL CONSEQUENCES OF SPENT FUEL STORAGE

The NRC concluded (NUREG 1984) that "there are no significant non-radiological consequences due to the extended storage of spent fuel which could adversely affect the environment." The construction of an interim spent fuel storage facility (i.e., the construction of a water pool, a concrete pad, a building, rail spur, etc.) would have little impact on the environment. The amount of heat given off by spent fuel decreases with time as the fuel ages and decays radioactively, and the amount of additional energy and water needed to maintain spent fuel storage is also small.

C.5 LAND UTILIZATION

With the use of water pool storage or dry container storage at an existing shipyard, land already devoted to industrial use is planned to be used for the spent fuel storage facility. The amount of land required for storage at specific shipyards is addressed in Attachment D.

C.6 COST

The use of alternate sites other than INEL would involve the construction of additional storage facilities. Both water pools and dry container storage could be used, with little environmental impact; therefore, the relative cost between these two options could be relevant. Conceptual cost estimates have been prepared for each storage option at each location that is being evaluated. These cost comparisons are found in Attachments D and E.

C.7 SUMMARY

Based on the above discussion, both a new water pool and dry container storage would be suitable for the interim storage of spent naval fuel with no important radiological or non-radiological environmental impact. If a facility would be required to be used for the inspection of spent fuel, as well as storage, then a water pool offers an advantage since water is an inexpensive and convenient form of transparent shielding. If it were not necessary for a new facility to be used to inspect spent fuel, then the cost of the facility and the amount of land required could be factors in selecting an option.

C.8 REFERENCES

MOCSG (The Midwestern Office of the Council of State Governments), 1993, *Report on Interim Storage of Spent Nuclear Fuel*, DOE/CH/10402--22, April.

NEA (Nuclear Energy Agency), 1993, *The Safety of the Nuclear Fuel Cycle*. Organization for Economic Co-Operation and Development.

NUREG (U.S. Nuclear Regulatory Commission), 1980, *Dry Storage of Spent Nuclear Fuel*, NUREG/CR-1223, April.

NUREG (U.S. Nuclear Regulatory Commission), 1984, "Waste Confidence Decision," *Federal Register*, Volume 49, No. 171, August 31.

ATTACHMENT D - DESCRIPTION OF STORAGE OF NAVAL SPENT NUCLEAR FUEL AT SERVICING LOCATIONS (SHIPYARDS AND PROTOTYPES)

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont)

LIST OF FIGURES

LIST OF TABLES

ATTACHMENT D

DESCRIPTION OF STORAGE OF NAVAL SPENT NUCLEAR FUEL AT SERVICING LOCATIONS (SHIPYARDS AND PROTOTYPES)

D.1 STORAGE OF NAVAL SPENT NUCLEAR FUEL IN CONTAINERS AT SHIPYARDS AND PROTOTYPES

D.1.1 Introduction

This attachment examines the alternative of storing naval spent nuclear fuel at shipyard and prototype sites where the fuel is removed from the reactor plant. Water pool storage, immobile dry storage containers, and dry storage in shipping containers are evaluated for each shipyard and prototype location. Under the No Action alternative, naval spent nuclear fuel would be stored in shipping containers. For the other alternatives where naval spent nuclear fuel would be stored at shipyard and prototype sites, the storage mode would be selected by the Record of Decision. Attachment C has addressed the generic safety of water pool and dry storage and concluded that both methods would be suitable for the interim storage of naval spent nuclear fuel with very little environmental impact. This attachment addresses the design requirements, operational considerations, costs, and land requirements for the Puget Sound Naval Shipyard, Pearl Harbor Naval Shipyard, Norfolk Naval Shipyard, Portsmouth Naval Shipyard, and the Kesselring Site.

The interim storage facilities for naval spent nuclear fuel at shipyards and prototype locations would be designed to comply with applicable requirements. The storage facilities would be monitored and maintained in compliance with Naval Reactors Program requirements for radiation protection of workers and the public and the environment. Specifically, exposure to workers at the storage site would be maintained as low as reasonably achievable and would be controlled to Naval Reactors Program radiation exposure standards. As with current naval practices, no measurable increase in radiation levels at the site boundary would result from the storage of naval spent nuclear fuel at any alternate site.

D.1.2 Shipping Containers

D.1.2.1 ***Container Design Features.*** Shipping containers and immobile dry storage containers position the spent naval fuel modules within sealed structures designed to physically constrain, support, and remove residual heat from the fuel in an environment that prevents corrosion of the fuel. The massive size of the containers provides not only strength, but also shielding against exposure to radiation from the spent fuel within.

The shipping containers might be M-140 shipping containers with long-lived seals suitable for storage of spent nuclear fuel for the duration of the period covered by this Environmental Impact Statement (EIS). A description of the M-140 shipping container is provided in Attachment A. This container is already certified to meet the requirements of the U.S. Nuclear Regulatory Commission, contained in 10CFR71, for the transportation of naval spent nuclear fuel. With installation of a long-lived seal, the M-140 container could be qualified for storage for 40 years. The shipping containers could either be positioned on railcars at the storage site or on concrete pads. The process of designing the shipping container long-lived seal would commence with the Record of Decision if this option were selected. The cost associated with the design and recertification of the shipping container would range from approximately $1 million to $5 million. The cost to manufacture each shipping container would be about $5 million. Some uncertainties in estimated costs exist due to the fact that a detailed design for the shipping container long-lived seal is not yet available.

If the Record of Decision were to choose shipping containers, a more detailed evaluation would need to be performed to determine whether it is more appropriate to modify the M-140 shipping container design or whether a new container design should be used. Since the M-140 was designed as a shipping container, the modifications that would need to be made to convert an M-140 to accommodate interim storage might involve substantial new design work and recertification for shipping.

About 500 additional containers with holding capacity equivalent to the M-140 container would need to be fabricated to cover the projected reactor servicing from 1995 through 2035. If an alternative using the shipping containers were to be chosen, an expanded manufacturing vendor base would need to be developed to meet the projected container requirements. With the current

manufacturing capabilities, 3 years are required to build an M-140 container and the output capacity is about 6 containers per year.

The shipping containers loaded during the period preceding the Record of Decision would also need to be modified to meet the storage container design criteria. An evaluation would be performed to determine whether these modifications could be safely made with spent nuclear fuel present in the containers. In the event that the spent nuclear fuel must be removed from the shipping containers, the containers would be unloaded and the spent nuclear fuel would be transferred into modified shipping containers at a suitable facility under controls which would protect workers, the public, and the environment. The unloading of spent nuclear fuel from the original shipping containers and reloading into modified shipping containers would introduce additional spent nuclear fuel handling, transportation, and risks.

D.1.2.2 ***Operations.*** The process of loading spent nuclear fuel into shipping containers for storage would be similar to that used for loading M-140 shipping containers. During reactor refueling operations, spent nuclear fuel is normally loaded into M-140 shipping containers that are filled with water. The spent nuclear fuel is staged in this configuration for sufficient time to ensure that heat produced by radioactive decay of fission products is adequately dissipated. When the water is removed from the M-140 container, the loaded M-140 can be shipped. After water is drained from the shipping container, it would be transported to the storage site. The water is processed for reuse. The transportation procedures would be essentially unchanged from current procedures except that containers would be moved to the interim storage site instead of being shipped to the Expended Core Facility (ECF) at the Idaho National Engineering Laboratory (INEL) for inspection. For railcar storage, the railcar would be positioned in the storage area. For cases where the shipping container is stored on a concrete pad, the container would be off-loaded from the railcar or truck, positioned, and then secured to the pad (if securing would be required). In order to accomplish this transfer, a large capacity crane would be needed at each site, and the site would need to be prepared as necessary to accommodate the mode of storage.

D.1.3 Immobile Dry Storage Containers

D.1.3.1 ***Container Design Features.*** There are currently no immobile dry storage containers designed for interim storage of naval spent nuclear fuel. The container design would be similar to that of containers which are presently certified by the Nuclear Regulatory Commission for storage of spent nuclear fuel from commercial reactors. The design, approval, and construction of an immobile dry storage container would commence with the Record of Decision if this option were selected. This effort could require up to 5 years to complete. The cost associated with the design and approval of the immobile storage container would be about $2 million. The cost to construct each immobile dry storage container would be about $2 million. These estimates are based on costs of commercially available containers with contingencies added to account for additional design features that may be required.

Two concepts for storing naval spent nuclear fuel in immobile dry storage containers have been developed in order to provide a baseline for assessing the impacts. Other dry storage approaches (such as dry storage vaults) exist and would be considered in more detail if the Record of Decision were to choose the immobile dry container storage alternative. The first approach (referred to as the minimum fuel loading concept) is based on the number of spent fuel assemblies stored in the immobile dry storage container being about the same as that which is loaded into M-140 shipping containers. This approach results in the need for about 500 immobile dry storage containers. The second approach (referred to as the maximum fuel loading concept) maximizes the number of fuel assemblies that would be stored in the immobile dry storage containers. The number of containers required for the second approach is about 300.

The minimum fuel loading concept results in a container with a comparatively simpler design, less maintenance, and lower unit costs (~$1.9 million/container). Under the maximum fuel loading concept, the container would need to be equipped with additional active cooling features such as water circulation to ensure that the heat produced by radioactive decay of fission products is adequately removed. These additional cooling features would be needed for a period of several years after the spent nuclear fuel is removed from the reactor vessel. For the minimum fuel loading concept, additional active cooling features such as recirculating water would not be required to remove heat. As with the shipping containers, an expanded vendor base would be necessary in order to construct the immobile dry storage containers at the rate they would be needed.

Figures D-1 and D-2 provide conceptual layouts of candidate immobile dry storage containers for naval spent nuclear fuel.

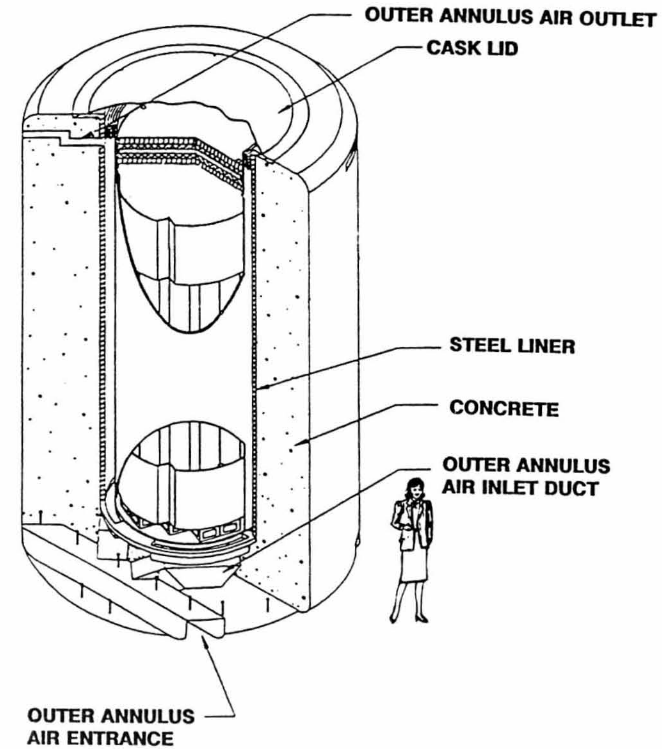

Figure D-1. Conceptual concrete immobil ·lry storage container for naval spent nuclear fuel.

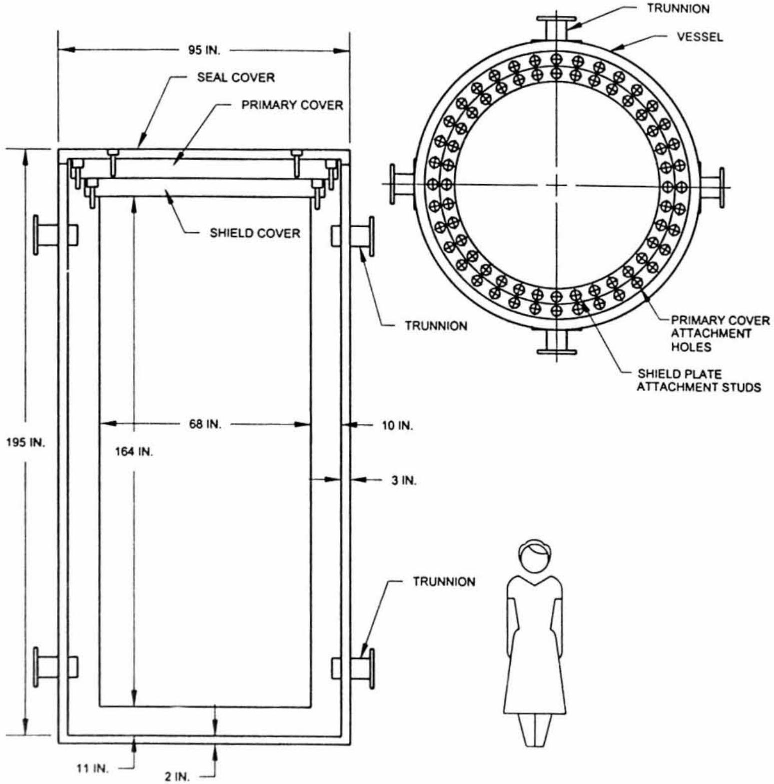

Figure D-2. Conceptual vertical metal immobile dry storage container for naval spent nuclear fuel.

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The dimensions of the immobile dry storage container that would be used for naval spent nuclear fuel would be approximately the same as the M-140 shipping container (i.e., approximately 10 to 16 feet high and 8 to 10 feet wide). The fuel spacing within the container and the container itself would be designed to prevent any nuclear chain reaction, to ensure that decay heat is adequately dissipated, and to ensure that the spent fuel would be protected from hazards associated with natural phenomena or human activities for each storage site.

D.1.3.2 Operations. Operations commence following the defueling of the reactor, after fuel modules are in a suitable holding container such as an M-130 or M-140 shipping container. The immobile dry storage container would be positioned at the storage location. Transfer of a spent fuel module from the holding container to the dry storage container would be accomplished one fuel module at a time using a shielded transfer container. All fuel transfers would be conducted in strict accordance with procedures which would have been written, reviewed, and approved by personnel trained, qualified, and specifically authorized to perform such work. The transfer container would be landed on the holding container, and a module would be withdrawn from the holding container. The module would be secured and the loaded transfer container closed, moved into position over the dry storage container, and landed. The transfer container would be reopened and the module lowered and seated in the immobile storage container. The transfer container would then be removed. This process would be repeated until the container is filled with spent fuel modules. The container would then be sealed.

Transfers of spent nuclear fuel to the immobile dry storage container would be conducted in accordance with Naval Reactors Program requirements for radiation protection. Radiological containment devices would be used where necessary to prevent radioactivity from spreading to the workplace and from becoming airborne. The transfer and storage containers would contain radiation shielding that minimizes radiation exposure to the workers during transfer and storage operations and ensures that radiation levels at the site perimeter are indistinguishable from natural background.

D.1.4 Water Pool Storage

D.1.4.1 Water Pool Design Features. If the Record of Decision were to choose the alternative of storing naval spent nuclear fuel in water pools, five water pools could be constructed, one at each designated storage site. Each water pool facility would be designed, built, and operated in accordance with DOE Order 6430.1A and consistent with the intent of Nuclear Regulatory Commission requirements in 10CFR72 and associated Regulatory Guides. The siting, design, construction, and approval of a water pool storage facility would commence with the Record of Decision and could take 6 to 9 years to complete. The design and construction of each water pool facility would also conform with local construction standards for each site.

Water pools operate by holding spent fuel modules in a deep pool of water. The water provides cooling for the spent fuel, a transparent medium for work activities, and protection from radiation (see Attachment C). The structural materials of the fuel modules and naval fuel cladding, as well as temperature and chemistry control of the water, would result in the spent fuel being highly resistant to corrosion. Corrosion-resistant racks below the water surface would be used to support and position the fuel modules in place for handling and to prevent a critical mass being formed. The water depth would be sufficient to provide shielding to protect workers and the environment during module movement and storage.

D.1.4.2 Operations. The naval spent nuclear fuel would be transferred to the water pool in a suitable container, such as an M-130 or M-140 shipping container. The fuel modules would then be transferred into the water pool using equipment and procedures that are similar to well-proven procedures used at ECF for unloading spent nuclear fuel from shipping containers. The spent nuclear fuel modules would be individually lowered and secured in the storage racks located on the water pool floor. The use of a water pool for storage of naval spent nuclear fuel would provide an opportunity for limited visual inspection of the exterior of the fuel modules after removing them from the naval vessels. This opportunity would not exist to the same extent for the dry storage container alternatives.

D.1.5 Design Basis Considerations for Storage Containers and Water Pools

The design of both the shipping and immobile dry storage containers would be in accordance with DOE Order 6430.1A and consistent with the intent of Nuclear Regulatory Commission requirements for independent spent fuel storage installations found in 10CFR72 and associated Regulatory Guides. Attachment F describes the exposures which would be expected during normal operational exposures and the exposures calculated for hypothetical accidents that might occur during interim storage of spent fuel at each shipyard and prototype location. The accidents that would be used to establish the requirements for the design of the interim storage facilities are discussed below.

D.1.5.1 Design Basis Considerations for Storage Containers.

(1) **Natural Phenomena.** The fuel spacing within the container and the container itself would be designed to prevent a nuclear criticality, to ensure that heat produced by radioactive decay of fission products is adequately dissipated, and to ensure that the container would safely survive hazards associated with natural phenomena such as storms or flooding for each storage site. The shipping containers and the immobile dry storage containers would be designed to withstand the most severe design basis seismic event expected for the storage sites. The seismic analysis would evaluate the internal and external structures of the containers and the components associated with stability of the containers. The containers and associated components would be designed to protect the environment during other natural phenomena such as tornado winds, tornado missiles, hurricanes, volcanic activity, design basis floods, and very large waves. If the Record of Decision involves the need for new facilities for the interim storage of naval spent nuclear fuel, detailed site-specific seismic evaluations would be conducted for those sites, and the results would be incorporated into the design of new facilities. The construction of any new facilities for naval spent nuclear fuel management would meet strict seismic standards for the interim storage of naval spent nuclear fuel. The design and construction of these facilities to seismic standards which take into consideration the seismic character of the area would ensure that structures could withstand a major seismic event. The adequacy of the storage facility would be documented in a safety assessment report for each location.

(2) **Man-made Hazards.** The containers would be arranged to allow access for routine inspections, maintenance, and emergencies. This includes sufficient accessibility for pressure, temperature, and radiological monitoring as well as for fire fighting equipment and ambulances.

The containers would be designed to withstand a fire without losing fission product containment. Flammable liquids and gases as well as explosive materials would be prohibited in the storage area with the exception of fuel in motor vehicles needed to support operations. Combustible materials such as wood, paper, and plastic would be kept to a minimum in the spent nuclear fuel storage areas.

The fuel spacing within the container and the container itself would be designed to prevent nuclear criticality, to ensure that the heat produced by radioactive decay is adequately dissipated, and to ensure that it would safely survive credible man-made accidents for each storage site. Other man-made hazards such as truck accidents, airplane crashes, and objects dropped by cranes would also be addressed in the safety assessment report.

D.1.5.2 Design Basis Considerations for Water Pools.

(1) **Natural Phenomena.** The spent nuclear fuel spacing within the water pool and the water pool itself and the building support structures would be designed to prevent criticality, to ensure that heat produced by radioactive decay is adequately dissipated, and to ensure that it would protect the fuel from the hazards associated with the design basis natural phenomena for each storage site (i.e., seismic, tornados, missiles generated by a tornado, hurricanes, volcanic activity, maximum expected floods, and very large waves). The water pools would be equipped with spent fuel storage racks for restraining the modules. The racks would be designed to safely survive the above hazards. If the Record of Decision involves the need for new facilities for the interim storage of naval spent nuclear fuel, detailed site-specific seismic evaluations would be conducted for those sites, and the results would be incorporated into the design of new facilities. The construction of any new facilities for naval spent nuclear fuel management would meet strict seismic standards for the interim storage of naval spent nuclear fuel. The design and construction of these facilities to seismic standards which take into consideration the

seismic character of the area would ensure that structures could withstand a major seismic event. The adequacy of the water pool facility would be documented in a safety assessment report for each location.

(2) **Man-made Hazards.** The water pool facility would be designed to withstand fire without damage to the spent fuel within the water. Flammable liquids and gases as well as explosive materials would be prohibited in the vicinity of the storage area with the exception of incidental quantities of flammable solvents necessary to support operations. Combustible materials such as wood, paper, and plastic would be kept to a minimum in the water pool facility.

The fuel spacing within the water pool would be designed to prevent criticality, and to ensure that it would safely survive credible man-made accidents for each storage site. Other man-made hazards such as truck accidents, airplane crashes, and crane drop accidents would also be addressed in the safety assessment report.

D.1.6 Shipyard and Prototype Locations

This section describes conceptual locations at the shipyard and prototype sites where storage facilities could be located to service refuelings and defuelings of naval ships. This section also lists land requirements for each storage method at each location, the construction cost for each method, and the associated operating cost.

D.1.6.1 Land Requirements. This section provides a summary of the land required for each of the storage methods at each of the locations where refueling and defueling are planned from 1995 through 2035.

These locations are the Portsmouth Naval Shipyard, the Puget Sound Naval Shipyard, the Pearl Harbor Naval Shipyard, the Norfolk Naval Shipyard, and the Kesselring Site. A map of each of these sites is provided in Figures D-3 through D-7, indicating a possible storage location at each of these facilities.

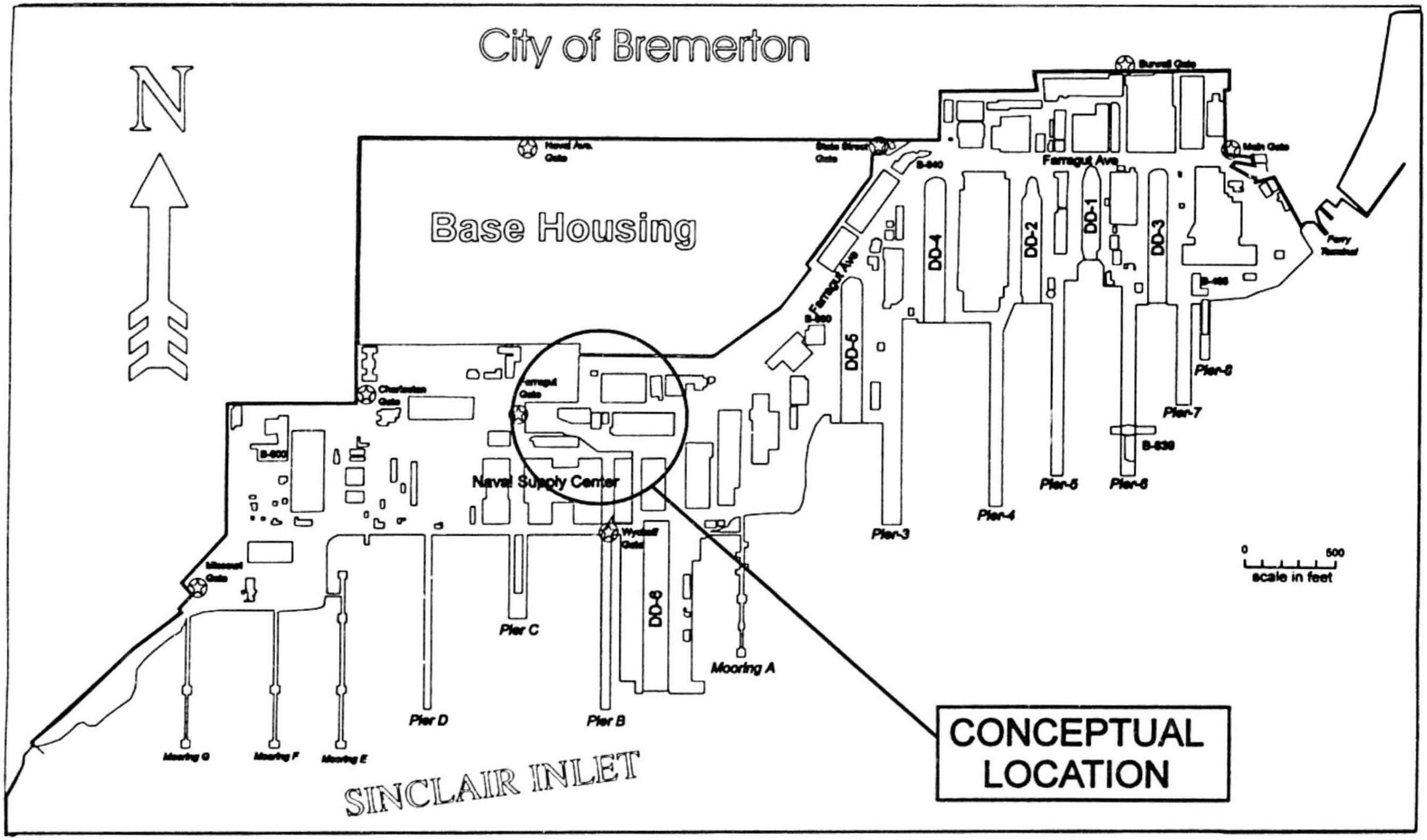

Figure D-3. Conceptual location of the interim storage site at Puget Sound Naval Shipyard.

139

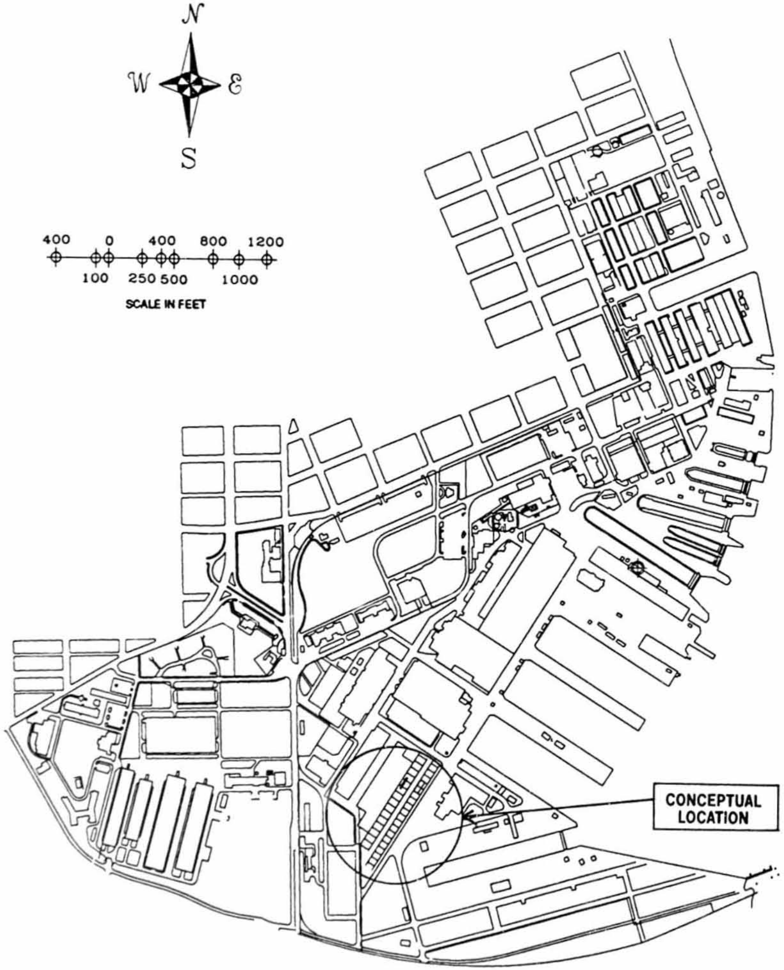

Figure D-4. Conceptual location of the interim storage site at Norfolk Naval Shipyard.

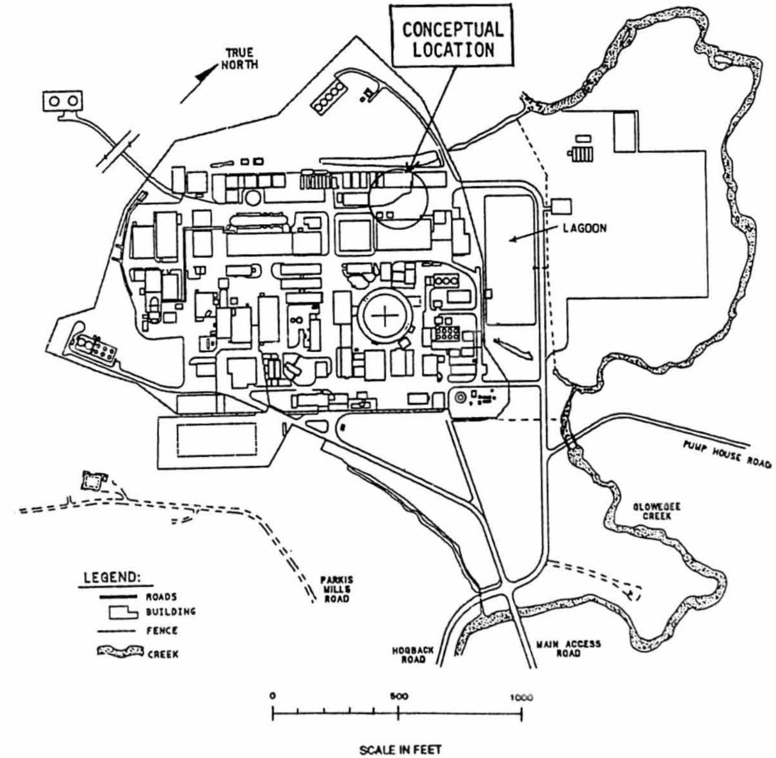

Figure D-5. Conceptual location of the interim storage site at Kesselring Prototype Site.

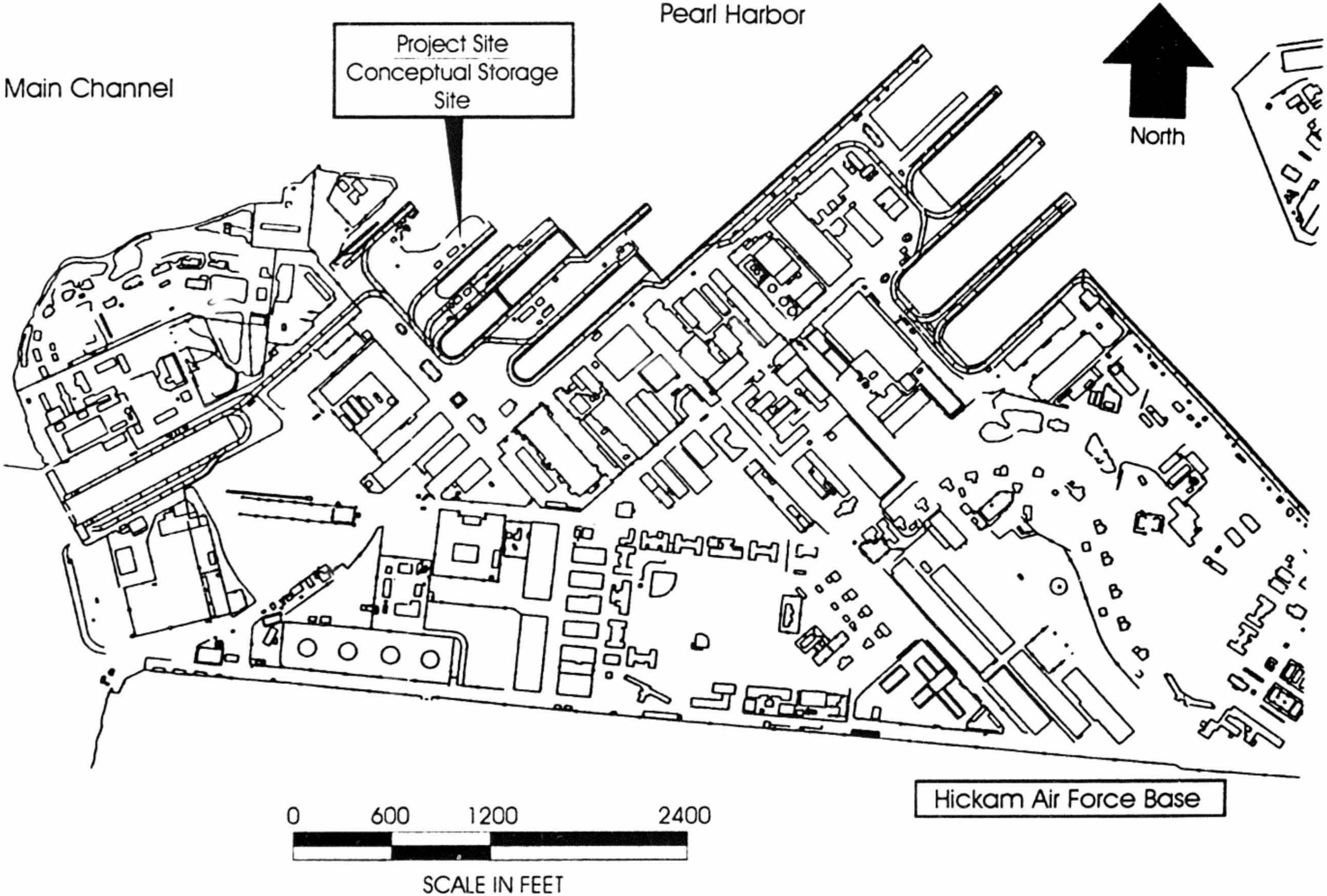

Figure D-6. Conceptual location of the interim storage site at Pearl Harbor Naval Shipyard.

142

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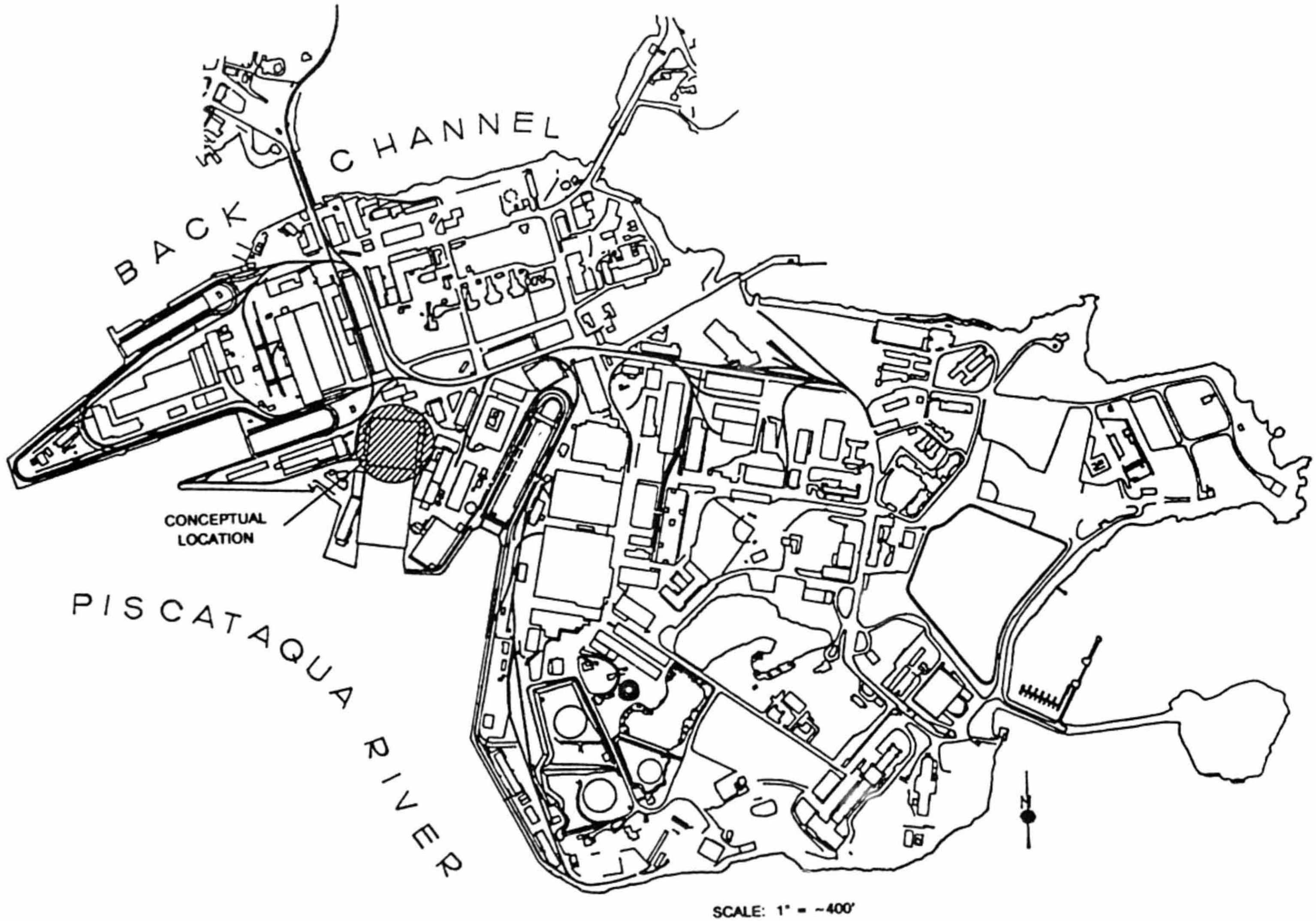


Figure D-7. Conceptual location of the interim storage site at Portsmouth Naval Shipyard.

143

Table D-1 provides a summary of the amount of land needed for each of the storage methods at each of the locations where storage of naval spent nuclear fuel could be located. It should be noted that the number of containers and land required could be slightly less than identified in Table D-1 as a result of actions taken during the transition period. As shown in Table D-1, storage utilizing shipping containers on railcars would typically require dedication of the most land.

Table D-1. Square feet of land required for storage facility.

Location	Number of Immobile Dry Storage Containers[1]	Number of Shipping Containers	Immobile Dry Storage Containers[2] (ft²)	Shipping Containers on Concrete Pad[3] (ft²)	Shipping Containers on Railcars (ft²)	Water Pool Facility[4] (ft²)
Portsmouth	27-51	61	10,000-19,000	18,000	72,000	20,000
Puget Sound	153-206	219	57,000-77,000	64,000	260,000	33,000
Pearl Harbor	21-30	42	8,000-11,000	12,000	50,000	20,000
Norfolk	132-219	247	49,000-82,000	72,000	293,000	31,000
Kesselring	5-6	6	1,900-2,000	1,700	7,100	17,000

(1) Range in required number of containers is due to options in conceptual design (see Section D.1.3.1).
(2) The immobile dry storage arrangement uses the containers stored on a concrete pad in double rows with one container diameter separation between adjacent containers. Each row is separated by a 15-foot wide accessway. Range in required land area is due to options in conceptual design.
(3) The shipping container arrangement uses the containers stored on a concrete pad in double rows with 4 feet between adjacent containers. Each row is separated by a 15-foot wide accessway.
(4) The water pool facility consists of a building that contains adequate space to house supporting equipment and facilities (approximately 17,000 ft²) and a water pool with adjacent work areas of sufficient size to accommodate the amount of spent nuclear fuel expected to be stored in the facility until 2035.

D.1.6.2 Site Construction, Container, and Operating Costs. This section provides estimated costs associated with each alternative for storing spent nuclear fuel at the shipyard and prototype sites. The major cost factors include facility construction or site preparation costs, container costs, and operating costs over the lifetime of the facility. Cost estimates are based on 1995 dollars.

Table D-2 provides a summary of the estimated construction costs for each storage option at each shipyard and prototype location. The construction costs for immobile and shipping containers on concrete pads and shipping containers on railcars include estimated costs for concrete (labor and materials), rails (for railcars), or cranes for lifting and handling containers or fuel transfer containers (for concrete pad storage). The majority of the construction costs for concrete pad storage options

Table D-2. Estimated site construction costs (millions of dollars).

Location	Immobile Dry Storage Containers on Concrete Pad	Shipping Containers on Concrete Pad	Shipping Containers on Railcars	Construction and Installation of Water Pools
Portsmouth	11-12	10	2	96
Puget Sound	15-16	13	5	141
Pearl Harbor	10-11	9	1	95
Norfolk	14-17	14	6	135
Kesselring	10	8	1[1]	89
Total	60-66	54	15	556

(1) Estimate does not include costs associated with establishing railroad extension from the access railroad to the storage site.

are associated with the need for a high-capacity crane. Water pool construction costs include estimates of costs for construction of the water pool, building structure, and associated support equipment. The table shows that construction costs for a water pool facility exceed those of other alternatives, and that shipping containers on railcars involves the lowest construction costs. However, the water pool facility construction costs represent a complete facility ready to hold spent nuclear fuel for interim storage. The construction costs in Table D-2 for the other storage modes represent completed site construction without the cost of the containers (see Table D-3) to hold the spent nuclear fuel.

Table D-3 provides a summary of the estimated costs to build shipping containers and immobile dry storage containers through 2035. The table shows that the immobile dry storage containers are the least expensive containers, and that the cost to build shipping containers to rest on concrete pads is slightly lower than to rest on railcars. The difference in cost between the two shipping container options is due to the cost of a dedicated railcar during storage. The shipping container costs in Table D-3 would be reduced by about 13 percent due to actions taken during the transition period (these actions are described in Section 3.8) to ship containers from the shipyards to ECF. Consequently, the total costs for shipping containers on concrete pads and shipping containers on railcars considering the transition period would be about 2615 and 2760 million dollars, respectively.

Table D-3. Estimated container cost (millions of dollars).

Location	Immobile Dry Storage Containers on Concrete Pad[1]	Shipping Containers on Concrete Pad	Shipping Containers on Railcars[2]
Portsmouth	55-100	319	337
Puget Sound	314-406	1145	1209
Pearl Harbor	43-59	220	232
Norfolk	271-431	1292	1363
Kesselring	10-12	31	33
Total	693-1008	3007	3174

[1]Range in container costs due to options in conceptual designs (see Sections D.1.2.1 and D.1.3.1). The lower end of the range represents container costs for the maximum fuel loading option (which requires fewer containers).
[2]Includes the cost of an equal number of railcars and containers required for this option.

Table D-4 provides the estimated costs to operate a naval spent nuclear fuel storage area. The operating costs include estimates of cost for personnel to monitor the facility, handle the spent nuclear fuel when it arrives at the facility, and maintain the facility. These estimates do not include the costs associated with eventual preparation of spent fuel for shipment to a site for disposition. Disposition preparation costs cannot be estimated at this time because the method for preparing the spent fuel has not been defined. Table D-4 shows that the lowest operating costs are associated with shipping containers on concrete pads and that water pool storage requires the highest operating costs.

Table D-4. Estimated operating costs through the year 2035 (millions of dollars).

Location	Immobile Dry Storage Containers on Concrete Pad	Shipping Containers on Concrete Pad	Shipping Containers on Railcars[2]	Water Pool
Portsmouth	11	3	8	180
Puget Sound	23	4	24	206
Pearl Harbor	11	3	6	180
Norfolk	21	4	27	206
Kesselring	9	2	3	124
Total	75	16	68	896[1]

[1]For comparison, the estimated operating cost (personnel to monitor and handle fuel and maintain the facility) for the ICPP Building 666 for the same period is 232 million dollars.
[2]Includes cost to replace or refurbish railcar after prolonged storage.

D.1.6.3 Total Construction and Operating Costs. Table D-5 is a compilation of the data contained in Tables D-1 through D-4, and calculated based on the entire 40-year period from the Record of Decision (1995 through 2035). This table shows that the total costs associated with the use of immobile dry storage containers are the lowest of all the storage options considered except for storage at Puget Sound and Norfolk where the largest amounts of spent fuel would be stored. In these cases, the total costs for using water pool storage are within the same range of approximation as immobile dry container storage.

Table D-5. Total costs through the year 2035 (millions of dollars).

Location	Immobile Dry Storage Containers on Concrete Pad(1)	Shipping Containers on Concrete Pad	Shipping Containers on Railcars	Water Pool
Portsmouth	77-123	332	347	276
Puget Sound	352-445	1162	1238	347
Pearl Harbor	64-81	232	239	275
Norfolk	306-469	1310	1396	341
Kesselring	29-31	41	37	213
Total Cost	828-1149	3077	3257	1452

(1)Range in total costs due to options in conceptual design (see Section D.1.3.1). The lower cost is associated with the maximum loading concept.

D.1.7 Time Required to Implement Each Storage Method

If the Record of Decision were to choose one of the alternatives involving storage of naval spent nuclear fuel at shipyards and prototype sites, some period of time would be required after the decision to fully implement the selected storage alternative. This section examines the time required to implement each storage method.

D.1.7.1 Container Storage. Implementation of the alternatives involving use of immobile dry storage containers and shipping containers could be viewed as a three-phase process. The first phase would cover the time required to design the container or container modification, to review and accept the design, to approve the container, to establish contracts for container fabrication, and fabricate the first container. During this phase, the shipyards and prototype sites where the containers would be stored would also construct or modify the container storage location as appropriate for the alternative chosen. For immobile dry storage containers, this phase would take about 5 years, if 2 years are required to design and accept the container design, 1 year is needed for approval of the container, and 2 years are required to build the container. For containers designed for both storage and shipping, this process would take about 5 years, based on 1 year to design the modifications, 1 year to approve the container, and 3 years to build the container.

The second phase would involve establishing funding. This will take approximately 3 years to complete. The third phase of the implementation period would involve fabrication of the remaining required containers. The estimate of the number of containers is based on the projected schedule for naval vessel refuelings and current estimates of the amount of spent nuclear fuel that would be placed into the containers. Although production rates for immobile dry storage containers and shipping containers are unknown, they can be approximated from existing production rates for M-140 shipping containers. With current manufacturing capabilities, 3 years are required to build an M-140 container, and the manufacturing capacity is about six containers per year. This production rate would need to be accelerated to 18 to 24 containers per year by increasing the number of manufacturers and by making fabrication process improvements. If the production rate of immobile dry storage containers and shipping containers is the same as that of M-140 containers and production rates can be increased as noted above, the supply of immobile dry storage or shipping containers would meet the demand for these containers at some point after the first several years. During the transition period, when an insufficient number of containers would be available to store all the spent fuel planned to be removed from U.S. Navy nuclear-powered vessels, some other means of storing naval spent nuclear fuel would be needed. As described in Section 3.8 of this EIS, it is expected that a transition period of 3 years of shipping followed by 3 years of allowing naval spent nuclear fuel to be stored in shipping containers at shipyards would provide the necessary storage space.

D.1.7.2 Water Pool Storage. If 6 to 9 years would be required to design, approve, and construct a water pool facility and this process would be initiated for each location within a year after the Record of Decision, water pools would be available for storage of naval spent nuclear fuel about 7 to 10 years following the Record of Decision. During the transition period, when water pools would be under construction at selected locations, some other means of spent nuclear fuel storage would be needed, such as the method described in Section 3.8.

D.1.8 Summary

Table D-6 summarizes the major advantages and disadvantages of the spent nuclear fuel storage alternatives previously discussed in this attachment.

Table D-6. Comparison of naval spent nuclear fuel storage alternatives.

Storage Mode	Advantages	Disadvantages
1. Shipping Container		
A. Storage on Railcars	1. Least amount of container handling after arrival at storage location. 2. Eliminates the need to remove spent fuel modules from the transfer container upon arrival at the storage site.	1. Railcars must be refurbished or replaced after prolonged storage. 2. Requires the largest land area of the storage options, except for Kesselring. 3. Shipping containers are more expensive than immobile dry storage containers and water pools (water pools cost more when small fuel quantities are stored such as at Kesselring).
B. Storage on Concrete Pads	1. Eliminates the need to remove spent fuel modules from the transfer container upon arrival at the storage site. 2. Concrete pads are less expensive than railcar storage if railcars must be replaced or refurbished.	1. More container handling required compared to railcar storage option (if containers will not need to be removed from railcar). 2. Higher total cost than immobile dry storage containers and water pools* (*when large quantities of fuel are stored).

Table D-6 (Cont).

Storage Mode	Advantages	Disadvantages
2. Immobile Dry Storage Containers	1. Lowest total costs of all the storage options.	1. The maximum fuel loading concept requires that the containers be filled with water for cooling purposes for several years after removal from the reactor. This requires additional maintenance and slightly increases risk of low-level contamination spillage during accidents. 2. Must remove spent fuel from transfer container and load it into immobile container.
3. Water Pool Storage	1. Has a lower total cost than shipping containers, except for Pearl Harbor and Kesselring which have less containers. 2. Provides opportunity for conducting visual examinations.	1. Has the highest operating costs of all the storage options. 2. Must remove spent fuel from transfer container and load into water pool.

D.2 INSPECT HIGH PRIORITY FUEL AT PUGET SOUND NAVAL SHIPYARD

D.2.1 Introduction

This section of the attachment discusses the alternative of inspecting a limited amount of naval spent nuclear fuel at Puget Sound Naval Shipyard (hereafter referred to as Puget Sound) to provide information on nuclear fuel performance for use in the development of advanced nuclear reactors. The inspections would be performed at the shipyard's existing Water Pit Facility. The limited amount of fuel inspected would be stored at Puget Sound following inspection, and all other spent fuel would be stored in a facility at or near the refueling or defueling sites until the time that permanent geologic storage becomes available.

D.2.2 Water Pit Facility Description

The Water Pit Facility is located at the west side of Dry Dock 5, within the industrial zone of Puget Sound. This zone consists of facilities involved in ship construction and repair, dry docking, and conversions. The area is bounded by Decatur Avenue on the north, the waterfront on the south, the Naval Supply Center on the west, and the main gate on the east. The Water Pit Facility is located approximately 411 meters (1350 feet) from the nearest shipyard public property boundary. Figure D-8 illustrates the layout of the Water Pit Facility.

The Water Pit Facility was originally constructed to provide the shipyard with the capability to refuel nuclear-powered aircraft carriers, with the work for the first such refueling at Puget Sound expected to commence in approximately 2006. To date, the facility water pool has been used for refueling equipment demonstrations and testing.

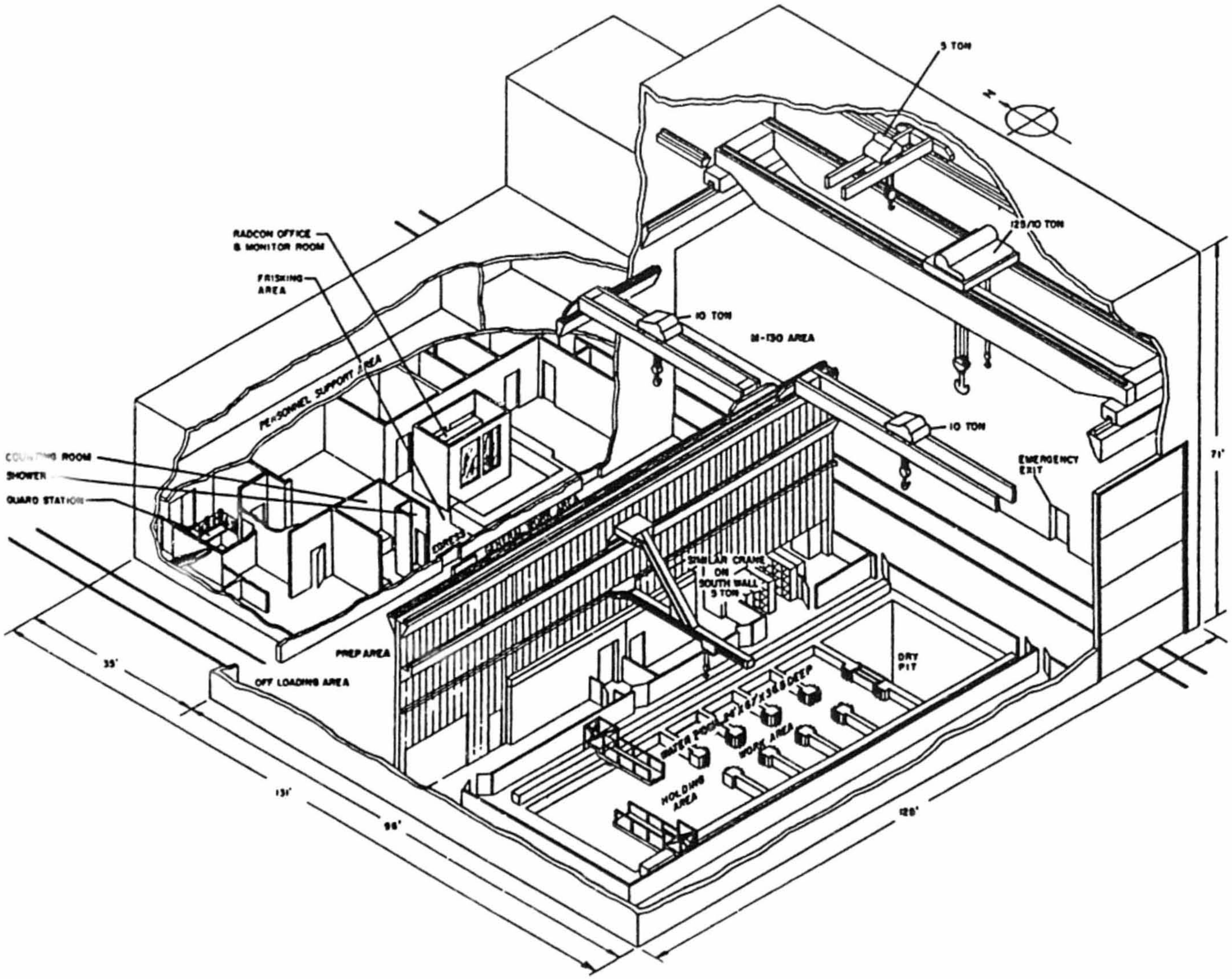


Figure D-8. Puget Sound Naval Shipyard Water Pit Facility.

The following key features of the Water Pit Facility are presented in terms of the facility's original aircraft-carrier refueling mission. Because of these design features, the facility is also considered suitable for limited naval spent fuel inspection operations.

1. A water pool for disassembly, assembly, and holding of fuel cells. The layout of the water pool is described below.

2. A work area for unpackaging, inspection, and preparation of new fuel clusters and associated equipment

3. An area for loading of shipping containers

4. A general use work area to support miscellaneous refueling support operations.

The Water Pit Facility is divided into two distinctive structures. The high bay structure is a radiologically controlled area containing the water pool and general work areas discussed above. This structure is designed to withstand the effects of design basis natural phenomena and of postulated failures of adjoining or adjacent structures without damage to the water pool or components in the water pool. The high bay walls are constructed of concrete to a height of 3.7 meters (12 feet) above ground level. The second structure is the Personnel Support Building which houses offices and other support areas. This structure is designed to meet the requirements of established naval facilities standardized criteria for structural design.

The water pool measures 7.3 meters (24 feet) wide x 20.4 meters (67 feet) long x 11.1 meters (36.5 feet) deep with a water depth of 10.5 meters (34.5 feet). It includes four work areas on each side of the pool at the east end to support refueling operations and a fuel holding area at the west end of the pool. Three of the four work areas are a nominal 2.1 meters (7 feet) x 2.1 meters (7 feet) and the fourth area is a nominal 2.6 meters (8.5 feet) x 2.1 meters (7 feet). The transfer aisle down the center of the pool is provided for all fuel and non-fuel movements. The water pool design includes provisions for isolation gates for each work area, for the fuel holding area, and for the dry pit. This isolation gate arrangement provides the capability to separate the various areas of the water pool if required. The dry pit, measuring 7.3 meters (24 feet) wide x 4.9 meters (16 feet) long x 11.1 meters (36.5 feet) deep, permits expansion of the water pool as needed.

D.2.3 Limited Inspection Operations

If future naval spent fuel examinations could not be accomplished at current capacity, the capacity which was available would be used to best advantage. Only naval spent nuclear fuel identified as having the greatest scientific value would be selected for detailed examination. Generally, this is spent nuclear fuel which is the first of a kind design or which has a characteristic of special interest.

Naval nuclear-powered ships would continue to be refueled and defueled at various shipyards across the country. Most of the spent fuel would be stored in a facility at or near the refueling and defueling sites until the time that permanent geologic storage becomes available. Those few fuel cells identified as high priority would be transported by railcar to Puget Sound in standard shielded shipping containers. Following its receipt in the Water Pit Facility's railcar work area, a shipping container would be prepared for fuel cell removal (dust cover removed, leveled, filled with water, containment installed, access plug removed). The fuel cells would be removed from the shipping container, one at a time, and transferred to the water pool in a shielded transfer container. The cells would be discharged into the pool and placed in the holding racks to await examination work. Upon completion of examination work, the spent fuel would be stored at Puget Sound as described in Section D.1. Storage facilities would have to be designed and certified to accommodate module sections resulting from spent fuel examinations as well as intact modules.

The following major items of water pool equipment (or equivalent) are considered necessary to support a high-priority naval spent nuclear fuel examination program. Also necessary are the relatively small and portable cameras and light sources for visual inspections. This equipment would support those spent fuel examinations currently performed in the ECF water pools at INEL as described in Section B.4.1 of Attachment B and summarized below.

EQUIPMENT ITEM	PURPOSE	FLOOR SPACE REQUIRED
Bandsaw/ Upender	Remove non-fuel structurals above & below fuel region to provide access for inspection and to rotate cells between vertical and horizontal orientations	46.4 m^2 (500 ft^2) 8.2 m x 5.6 m (27 ft x 18.5 ft)
Universal Inspection Station	Measure fuel cell dimensions	7.5 m^2 (81 ft^2) 2.7 m x 2.7 m (9 ft x 9 ft)
Vertical Inspection Gage	Trace contour of surfaces of fuel cell assemblies and control rods	16.7 m^2 (180 ft^2) 3.0 m x 5.5 m (10 ft x 18 ft)
Milling Machine	Section fuel cells into subassemblies, preassemblies, and elements for other examinations	11.1 m^2 (120 ft^2) 3.7 m x 3.0 m (12 ft x 10 ft)

Based on floor space requirements, the Water Pit Facility water pool and dry pit could not accommodate spent nuclear fuel examinations without removal of work area partition walls and without removal of the aircraft carrier refueling equipment. As a result, Puget Sound would no longer have the capability to refuel nuclear-powered aircraft carriers. Expansion of the Water Pit Facility to accommodate simultaneous refueling and examination operations is undesirable due to the proximity of other shipyard facilities.

Puget Sound does not have a shielded cell examination capability. Two options were considered for implementing such a capability:

1. Transfer fuel sections from Puget Sound to a shielded cell facility at another Naval Reactors site such as the Knolls Atomic Power Laboratory near Schenectady, New York, or the Bettis Atomic Power Laboratory near Pittsburgh, Pennsylvania. This would require additional shipments of spent fuel sections across the country. The spent fuel would be transported in shipping casks which would have to be certified for this purpose.

2. Construct shielded cells at Puget Sound. These cells would necessarily be sited some distance from the Water Pit Facility since sufficient space is not available either within the facility or adjacent to it in the industrial zone of the shipyard. In addition, a means

of transferring items for examination between the water pool and the shielded cells would have to be implemented. Shielded cask movements via truck and cart movements via underground tunnel are two possible means of transfer. This option is undesirable because it involves construction of a new facility but does not provide direct communication between the water pool and shielded cells.

Based on the above discussion, the alternative of examining a limited amount of naval spent nuclear fuel would include a full range of water pool visual and dimensional inspections at the Puget Sound Water Pit Facility and a full range of shielded cell examinations at another Naval Reactors site. This alternative would therefore include all INEL-ECF capabilities as described in Sections B.4.1 and B.4.3 of Attachment B.

D.2.4 Advantages and Disadvantages of this Alternative

Advantages

1. Portions of the naval spent nuclear fuel examination program could be moved from INEL-ECF without having to construct new facilities. A full range of water pool inspections could be accomplished at Puget Sound. A full range of shielded cell examinations could be accomplished at another Naval Reactors site.

Disadvantages

1. The small size of the water pool complicates placement of inspection equipment. As a result, the equipment would be limited in nature and would require removal of water pool work area partition walls and removal of aircraft carrier refueling equipment. As a result, Puget Sound would no longer have the capability to refuel nuclear-powered aircraft carriers.

2. Transferring items for examination between the water pool and shielded cells would involve additional spent fuel shipments across the country and would require design and certification of a container for this purpose.

D.2.5 Facility Support Systems

The systems which were intended to support the aircraft carrier refuelings will also support the limited naval spent fuel inspection efforts. These include the water pool fluid systems, the heating and ventilation systems, and the normal and emergency electrical power systems.

D.2.6 Radiation Sources

The primary sources of radiation in the Water Pit Facility would be the spent fuel and the associated irradiated components which are handled during inspection operations. Radiation results from the fission products which reside in the fuel region of the depleted clusters and are contained by the fuel cladding. The cladding around the fuel region would not be penetrated by any fuel cell cutting or sectioning operation in the Water Pit Facility. Irradiated non-fuel components are also sources of radiation, as are corrosion products which reside on all external surfaces. Handling operations could cause some of the corrosion products to become detached from the surfaces. Therefore, in addition to direct radiation, contamination must be considered in the control of radiation sources.

The water pool water is treated by the filtration and purification system to maintain the waterborne radioactivity as low as reasonably achievable, typically less than 1×10^{-6} microcurie Co-60/ml. This level of activity is below the concentration limit in 10CFR20, Attachment B, Table 2 for liquid effluents released to the general environment. The vessels and piping in the filter system then become potential radiation sources. The water must be considered a source even though its radiation level will be very low. The waterborne radioactive material causes equipment in the pools to become radiation sources, the water pool floor to become contaminated, and a radioactive scum ring to form on the walls of the water pool at the water surface. Even considering all of these sources contributing to the ambient radiation level in the water pool area, the controls which are exercised will ensure that the overall source is minimal and the occupational exposure remains as low as reasonably achievable.

There would normally be no airborne radioactivity generated by the handling of the cells in the water pool. However, very low levels of airborne activity (approximately 1×10^{-12} microcurie Co-60/ml) have been detected near the surfaces of other water pools. This level of activity is below

the concentration limit in 10CFR20, Attachment B, Table 2 for airborne effluents released to the general environment. The presence of even low-level airborne contamination will eventually lead to the ventilation system ductwork and HEPA filters becoming sources of radiation. This would occur over a very long period of time and the radiation levels would be controlled to a very low level. As noted above, the controls which are exercised will ensure that the occupational exposure remains as low as reasonably achievable.

D.2.7 Radiological Protection Features

The facility is designed to protect workers and the general public from radiological risk. Controls are such that workers receive much less than the allowable limits for radiation and radioactivity. The ventilation system is designed to mitigate the consequences of an accidental release of radionuclides within the Water Pit Facility building and to limit the atmospheric release at the stack. The double-walled (reinforced concrete, stainless steel liner) water pool is designed to prevent leakage under design earthquake force loading conditions. The radioactive fluid systems will maintain zero liquid discharge to the environment during Water Pit Facility operations.

D.2.8 Estimated On-Site Dose Assessment

The occupational radiation exposure for workers performing limited spent fuel inspections in the Water Pit Facility is expected to be consistent with that of ECF workers performing similar operations at INEL. As discussed in Section 5.2.12.1, radiation exposures to ECF workers at INEL have averaged approximately 100 mrem per year. The person-rem per year for the Water Pit Facility will vary with the manning level which is dependent on the spent fuel inspection activity occurring in the facility. However, the maximum manning level is anticipated not to exceed 60 people.

D.2.9 Seismic Design

Structural loadings due to seismic activity were determined as follows. Building floor response spectra for the horizontal and vertical directions were obtained from a three-dimensional damping mass spring model of the high bay which included soil-structure interaction, subjected to a 0.35 g ground acceleration value resulting from the seismic design analysis. The high bay superstructure and substructure were analyzed using the floor response spectra in separate finite element computer models. The superstructure model was subjected to structural loads which included a 113.5-metric ton (125-ton) load lifted by the large overhead crane. The combined forces of these loads with the seismic loads were applied to the substructure model at the column base plate locations. The substructure model was subjected to the design earthquake response spectra. This method was repeated for other combinations of structural loads with wind or tornado loads. Members were checked and designed for the maximum stress from any of the loading combinations. In addition, the water pool is designed to contain the pool water under design earthquake force loading conditions.

ATTACHMENT E - DESCRIPTION OF RECEIPT, HANDLING, AND EXAMINATION OF NAVAL SPENT NUCLEAR FUEL AT ALTERNATE DOE FACILITIES

TABLE OF CONTENTS

LIST OF FIGURES

ATTACHMENT E

DESCRIPTION OF RECEIPT, HANDLING, AND EXAMINATION OF NAVAL SPENT NUCLEAR FUEL AT ALTERNATE DOE FACILITIES

E.1 DISCUSSION

This attachment describes the options for establishing new or modified facilities that essentially duplicate the capabilities of the existing Expended Core Facility (ECF) at the Idaho National Engineering Laboratory (INEL). Also discussed herein are the differences from the existing facility, which is described in detail in Attachment B.

The capabilities of the ECF at INEL include detailed examinations of spent nuclear fuel from naval reactors and test specimens from the Advanced Test Reactor (ATR) at the INEL Test Reactor Area. It would be possible to provide ECF capabilities at an alternate DOE facility (Savannah River Site, Hanford Site, Oak Ridge Reservation, or Nevada Test Site) by constructing an entirely new facility. At Savannah River or Hanford, ECF capabilities could also be provided by modifying an existing facility. The preferred locations for siting an ECF at Savannah River, Hanford, Oak Ridge, and the Nevada Test Site are described in Sections 4.3.1, 4.4.1, 4.5.1, and 4.6.1, respectively. The main advantage of new construction is that the facility can provide all capabilities currently available at the ECF at INEL without limitations. The new construction water pool and shielded cell complex would be constructed in such a manner as to duplicate, as much as possible, the capabilities of the ECF at INEL. The existing ECF is highly capable, having been designed to accomplish the tasks required by the Naval Nuclear Propulsion Program. Key disadvantages of new construction, however, are high cost and the time necessary to initiate and complete construction.

Modification of an existing facility at Savannah River or Hanford which has at least some of the features that are required in a functional ECF would enable reductions in cost and time to achieve full capability, depending on how many facility modifications are required. A disadvantage, however, is that some of the methods currently in use at the ECF at INEL may also require modification to effectively and promptly utilize an existing facility, and such modifications may compromise the

capabilities of the examination facility. The existing facility that can be made a part of the Savannah River Site is the Barnwell Nuclear Fuel Plant (hereafter referred to as the Barnwell Plant) which is unused and available following acquisition from its present private corporate owners. The existing facility on the Hanford Site is the Fuels and Materials Examination Facility (FMEF) which is unused and available immediately. Sections E.2 and E.3 describe the modifications to existing facilities or to current processes that would be needed to provide the complete range of ECF capabilities at the Barnwell Plant and the FMEF. Section E.4 provides a discussion of how naval spent fuel and test specimen examination work would proceed through the interim period as this work is being transferred from the ECF at INEL to the ECF location at the alternate DOE facility.

Receipt and handling of naval spent fuel at the new ECF location at the alternate DOE facility would be similar to receipt and handling of spent fuel at the ECF at INEL as described in Section B.2 of Attachment B. Following all examinations at the new ECF, most of the spent fuel would be loaded in the water pool into shipping casks for transport to the long-term fuel storage location at the same DOE facility. The spent fuel would remain at this location until the time that ultimate disposition is possible.

The new ECF would also duplicate the capabilities of the ECF at INEL with respect to the assembly, disassembly, and examination of ATR irradiation test specimens.

E.2 USE OF THE BARNWELL PLANT AT SAVANNAH RIVER FOR ECF WORK

The Barnwell Plant is not owned by DOE but could be acquired and incorporated into the Savannah River Site property. It has a water pool complex with about 433 square meters (4660 square feet) of surface area (see Figure E-1) that can be utilized with minor modifications to perform unloading of naval fuel transport casks in a manner virtually identical to that employed at the ECF at INEL. An overhead crane running the length of the water pool would have to be added. However, providing naval spent nuclear fuel and test specimen examination capabilities comparable to the ECF at INEL would entail an expansion of the Barnwell Plant water pool to at least two times its present size. The design of the Barnwell Plant facility provides for such an expansion in an easterly direction while the existing water pool remains functional in a reduced capacity mode.

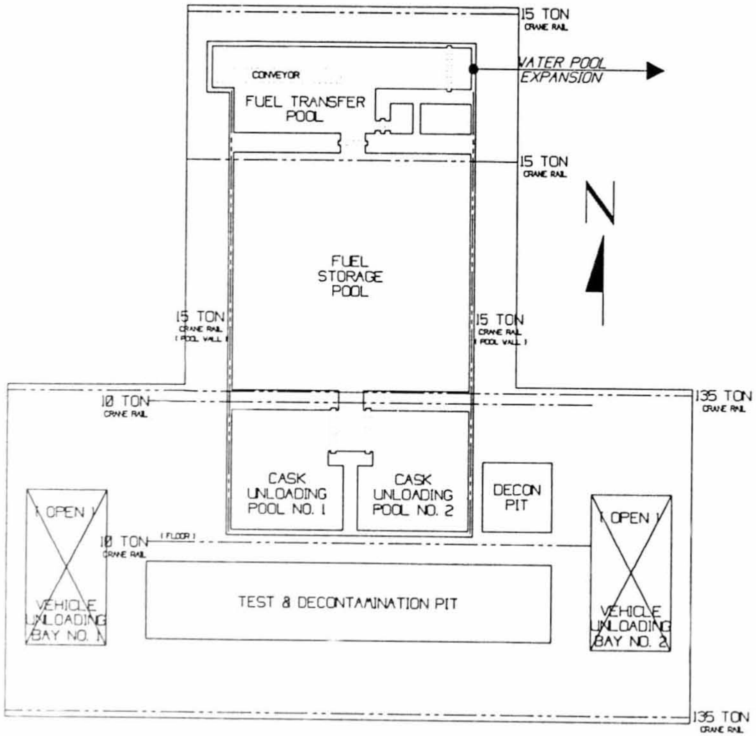


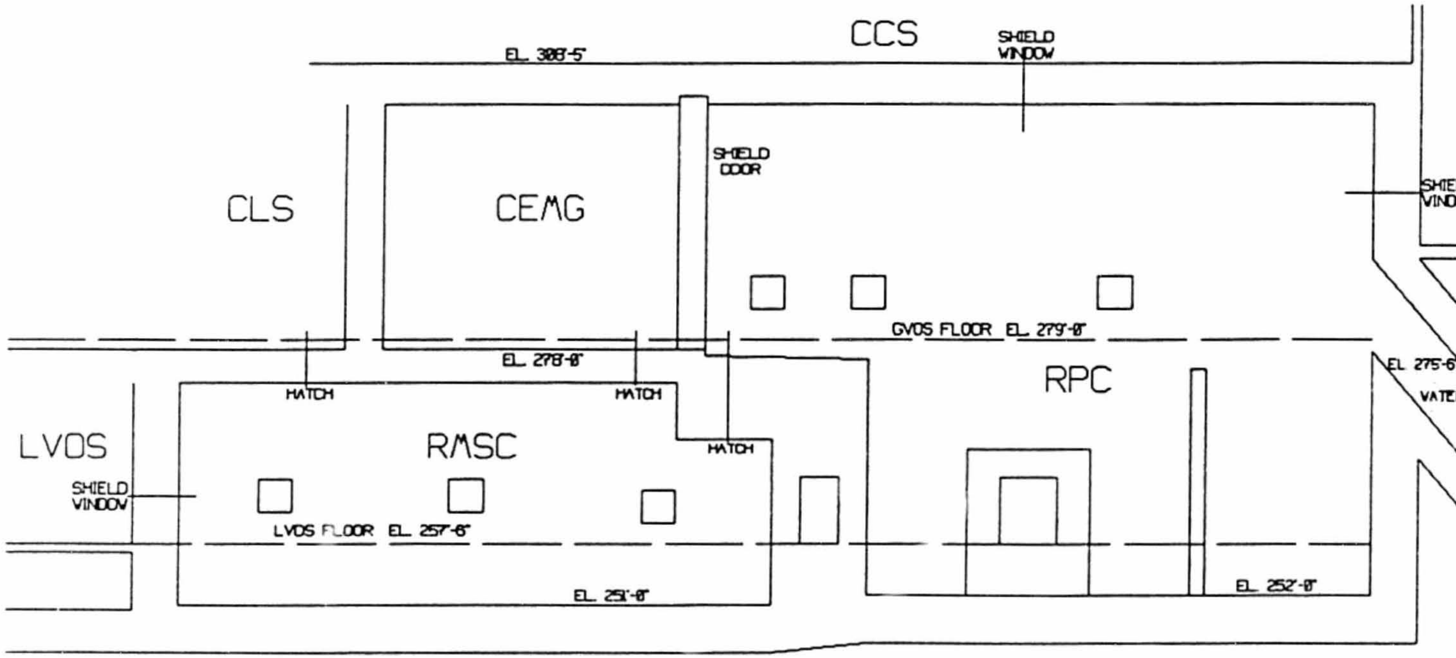
Figure E-1. Plan view of the Barnwell Plant Fuel Receiving and Storage Station.

It is envisioned that the full ECF shielded cell capabilities could be provided at the Barnwell Plant using a combination of the three remote maintenance cells and the eight sample and analytical cells. Material would be transferred from the water pool to the remote maintenance cells via a conveyor. The crane equipment maintenance gallery and the upper level of the remote process cell are connected by a shielded door; these cells are connected to the remote maintenance and scrap cell below by hatches (see Figure E-2). Additional work stations (viewing window and manipulator ports) would have to be added to service these cells. The remote maintenance cells are connected to the sample and analytical cells above via a waste chute which would have to be upgraded to improve transfer capability between these cell areas. Methods would have to be developed for material movement from one shielded cell elevation to another. The combined length of the ECF shielded cells at INEL is less than 57.9 meters (190 feet). The combined length of the Barnwell Plant remote maintenance cells and sample and analytical cells is greater than 67.1 meters (220 feet), so that sufficient cell work space should be available. There are also five contact maintenance cells available, although at present they have no workstations and are not connected to each other, to any other cell area, or to the water pool. An alternative to the Barnwell Plant water pool expansion would be to use the contact maintenance cells for some of the operations presently performed in the ECF water pool at INEL. Varying amounts of existing equipment and piping in the Barnwell Plant shielded cells would have to be removed and disposed.

Once modified, the Barnwell Plant would provide the full range of water pool and shielded cell examination capabilities. However, the arrangement of the cells in the fuel handling area could make material movement within the facility more difficult than material movement at the ECF at INEL. As a result, throughput in the Barnwell Plant could be adversely affected.

E.3 USE OF THE FUELS AND MATERIALS EXAMINATION FACILITY AT HANFORD FOR ECF WORK

The FMEF on the DOE Hanford Site in Washington currently has a large shielded cell complex that is suitable for ECF-type shielded cell operations with several modifications. Those modifications primarily entail the logistics associated with installing the equipment in the cells and transporting items for examination to and from this equipment.



RPC = REMOTE PROCESS CELL
RMSC = REMOTE MAINTENANCE AND SCRAP CELL
CEMG = CRANE EQUIPMENT MAINTENANCE GALLERY
CLS = CASK LOADING STATION
GVOS = GRADE VIEWING AND OPERATING STATION
LVOS = LOWER VIEWING AND OPERATING STATION
CCS = COLD CHEMICAL STATION

Figure E-2. Elevation looking north in the Barnwell Plant fuel handling area.

166

At present, there is no water pool at FMEF. One means of providing this portion of ECF capabilities would be to establish a dry cell facility. The FMEF main process cell, decontamination cell, and upper process cell were evaluated for such a facility (see Figure E-3). Conceptually, material would be transferred from shielded casks in the shipping and receiving crane bay into the decontamination cell via a ceiling port. At present, there are only small penetrations between the decontamination cell and main process cell; this would have to be upgraded to facilitate material transfer. The combined surface area of the three cells is about 706 square meters (7600 square feet), compared to at least 866 square meters (9320 square feet) for the conceptual expanded Barnwell Plant water pool discussed previously. This suggests that the full ECF water pool capabilities could not be provided in the dry cell facility. In addition, one or more of the process cells is intended for inclusion in the shielded cell complex (see next paragraph). Removal of decay heat from spent fuel and irradiation test specimens in temporary dry storage would have to be evaluated. It is concluded that duplication of ECF spent fuel and test specimen examination capabilities at FMEF would require construction of a new water pool at least two times the present size of the Barnwell Plant water pool. The location of the pool and the means for transferring items between the pool and the shielded cell complex would have to be evaluated.

It is envisioned that the full ECF shielded cell capabilities could be provided at FMEF using a combination of the main process cell and the 14 process support cells. The main process cell is connected to the process support cells below by hatches (see Figure E-3). There appear to be sufficient workstations (viewing window and manipulator ports) servicing all cells. Methods would have to be developed for material movement from one shielded cell elevation to the other. The combined length of the FMEF main process cell and process support cells is greater than 76.2 meters (250 feet), so that sufficient cell work space should be available. The decontamination cell and upper process cell would be available in support of shielded cell operations. The FMEF shielded cells are essentially empty.

Once modified, the FMEF would provide the full range of water pool and shielded cell examination capabilities. However, the arrangement of the cells in the fuel handling area and the separation of the water pool and shielded cells would make material movement within the facility more difficult than material movement at the ECF at INEL. As a result, throughput in the FMEF could be adversely affected.

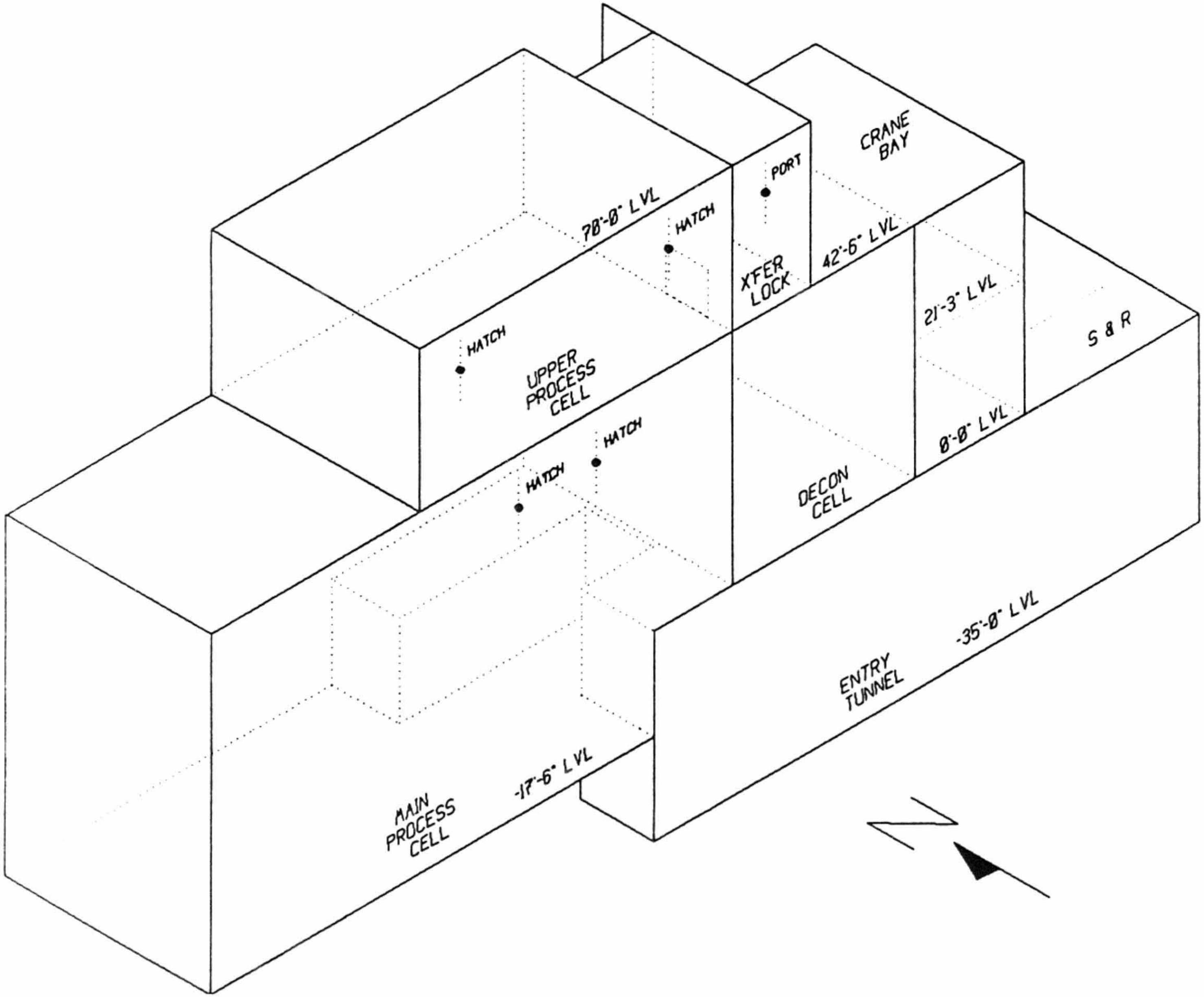


Figure E-3. FMEF fuel handling area.

108

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E.4 INTERIM OPERATIONAL PERIOD

A transitional period will exist between the date that the Record of Decision is issued and the date that the alternative selected can be fully implemented (unless the selected alternative maintains ECF operations at INEL). This transition period would be approximately 6 years. If it is desired that all ECF work be completely transferred to an alternate DOE facility, then actions would have to be taken to minimize the disruption in examination capability for naval spent nuclear fuel and ATR test specimens. This section discusses how this will be accomplished if the alternate DOE facility option is selected in the Record of Decision.

The Barnwell Plant would have to be acquired by the DOE from its present private corporate owners. It is estimated that less than $800 million in acquisition, modification, and construction costs would complete the Barnwell Plant for ECF usage.

The FMEF at Hanford is already owned by the DOE but it appears to require a greater amount of design effort to be a fully functional ECF since a large water pool would need to be constructed and tied in to the shielded cell complex in order to initiate fuel receipt. It is estimated that less than $800 million in modification and construction costs would complete the FMEF for ECF usage.

During the transitional period between the Record of Decision and full implementation of the selected alternative, shipments of naval spent nuclear fuel to the ECF at INEL would continue, pending construction of storage and examination facilities at the new site. All naval spent nuclear fuel would then be transferred to the new site.

ATTACHMENT F - ANALYSIS OF NORMAL OPERATIONS AND ACCIDENT CONDITIONS

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont)

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

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ATTACHMENT F

ANALYSIS OF NORMAL OPERATIONS AND ACCIDENT CONDITIONS

This attachment presents estimated environmental consequences, event probabilities, and risk (a product of probability and consequence) for both normal operations and postulated accident scenarios related to the storage and examination of naval spent nuclear fuel. Normal operations and accidents are evaluated to estimate the potential for releases of both radioactive material and toxic chemicals. The results of these analyses are presented in terms of the health effects to facility workers and the public predicted due to the release of radioactive materials and toxic chemicals into the environment. Effects on environmental factors are also presented, based on the amount of land which could be impacted due to postulated accidents.

Analysis results are presented for several different Department of Energy (DOE) and naval shipyard locations which are being considered as alternative sites for future naval spent nuclear fuel storage and examination. The DOE facilities evaluated include the Idaho National Engineering Laboratory (INEL), Savannah River Site, Hanford Site, Nevada Test Site, Oak Ridge Reservation (hereafter referred to as Oak Ridge), and Kenneth A. Kesselring Site. Puget Sound Naval Shipyard, Pearl Harbor Naval Shipyard, Norfolk Naval Shipyard, and Portsmouth Naval Shipyard have also been evaluated for naval spent nuclear fuel operations.

SUMMARY

Analyses of normal operations and design basis and beyond design basis hypothetical accidents were performed to estimate the potential consequences due to release of radioactive materials and toxic chemicals. The analysis results for radiological operations have been summarized by the locations and alternatives being considered in the Environmental Impact Statement.

Historical Accidents

The Naval Nuclear Propulsion Program has an outstanding nuclear safety record. In over 4500 reactor-years of operation and more than 300 refuelings and defuelings of Naval reactors, there

has never been a nuclear reactor accident, criticality accident, transportation accident, or any release of radioactivity having a significant effect on the environment.

Summary of Naval Spent Nuclear Fuel (SNF) Alternatives

Alternative	Description of SNF Activity
No Action	SNF retained at shipyards and Kesselring. Dry storage in containers only.
Decentralization No Examination	SNF retained at shipyards and Kesselring. Either dry containers or water pool storage would be used.
Decentralization Limited Examination	SNF retained at shipyards and Kesselring. Either dry containers or water pool storage would be used. Limited SNF shipments to Puget Sound Naval Shipyard for examination.
Decentralization Full Examination	All SNF shipped to INEL-ECF for examination. All SNF returned to origin for storage in either dry containers or water pools.
Planning Basis	SNF would be received, examined, and stored at INEL as in past years. The proposed dry cell facility would be completed at ECF.
Regionalization or Centralization	SNF would be received, examined, and stored at either INEL, Hanford, Savannah River, Nevada Test Site, or Oak Ridge.

Normal Operations

Table F-1 presents the estimated number of fatal cancers per year to the general population living within a 50-mile radius of each facility due to radiological releases from normal operations. The results in this table were calculated using the methods described in Section F.1.3. The number of fatal cancers is very low at all locations and for all alternatives.

The ISC2 computer code (EPA 1992b) was used to estimate the concentration of chemicals released during normal operations. The results show that for INEL, Hanford, Savannah River, the Nevada Test Site, the Barnwell Plant, and Oak Ridge, no ambient air quality standards would be exceeded; therefore, no adverse effects are expected. Heating boilers and emergency diesel generators already exist at the Navy shipyard locations and thus selection of these alternate locations would not result in a measurable increase in emissions.

Hypothetical Accident Evaluations

Several hypothetical accidents were analyzed at each facility for each of the alternatives. The results are summarized in Tables F-2 and F-3. The results in these tables were calculated using the methods described in Section F.1.3. Both fatal cancers from the maximum foreseeable accident at each location and the most severe risk from a facility accident at each location are presented. Risk is defined as the product of the consequences of an event multiplied by the probability of that event. The risks associated with the accidents analyzed have not been added together in order to avoid creating the impression that all risks have been calculated. The risks presented in this appendix cover the complete range of accidents which might make a detectable contribution to overall risk and additional analyses would not be expected to result in increases in calculated risk. The facility accident which results in the highest risk is a drained water pool at INEL, Hanford, Puget Sound, Portsmouth, and Kesselring. For Savannah River, Pearl Harbor, Norfolk, the Nevada Test Site, and Oak Ridge, an airplane crash into a dry storage area or a dry cell facility results in the greatest risk. As was the case for the normal operations evaluation, the accident risk is very low at all locations and for all alternatives.

Table F-4 presents a summary of the risk of fatal cancers by alternative for normal operations and most severe facility accident for each alternative. Consistent with the detailed tables, this summary table shows that all alternatives and all locations associated with spent fuel examination have very low risk.

Tables F-5 through F-8 present a summary by alternative of the impacts from all naval spent nuclear fuel facility radiological accidents which were analyzed.

A shipping accident in Puget Sound, at a location in the shipping lane approximately 2 miles from Seattle, was also analyzed using the methods described in this Attachment. This hypothetical accident results in a fire onboard the ship which involves spent nuclear fuel shipping containers. When compared to the facility accidents analyzed at Puget Sound Naval Shipyard, this shipping accident has a slightly lower risk of fatal cancers than the most severe facility accident at the shipyard.

The EPI computer code (Homann 1988) was used to estimate the concentration of chemicals released in the event of two postulated accident conditions. One postulated accident involved a chemical spill and fire at ECF and the alternate DOE sites and the other postulated accident involved a diesel fuel fire at ECF, the alternate DOE sites, and the shipyard locations. The chemical

Table F-1. Number of fatal cancers per year from normal operations (fatalities per year to general population located within 50-mile radius of site).

		No Action	Decentralization-No Examination	Decentralization-Puget Sound Exam	Decentralization-INEL Exam	Planning Basis/ Regionalization/ Centralization-INEL	Regionalization/ Centralization-Hanford	Regionalization/ Centralization-Savannah River	Regionalization/ Centralization-Nevada Test Site	Regionalization/ Centralization-Oak Ridge
DRY STORAGE AT NAVAL NUCLEAR PROPULSION PROGRAM SITES, WATER POOL STORAGE AT DOE SITES										
INEL		0.00	0.00	0.00	8.50×10^{-7}	8.50×10^{-7}	0.00	0.00	0.00	0.00
Hanford		0.00	0.00	0.00	0.00	0.00	4.00×10^{-6}	0.00	0.00	0.00
Savannah River		0.00	0.00	0.00	0.00	0.00	0.00	1.80×10^{-5}	0.00	0.00
Nevada Test Site		0.00	0.00	0.00	0.00	0.00	0.00	0.00	9.00×10^{-8}	0.00
Oak Ridge		0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	5.00×10^{-5}
Puget Sound		1.20×10^{-6}	1.20×10^{-6}	6.62×10^{-5}**	1.20×10^{-6}	0.00	0.00	0.00	0.00	0.00
Pearl Harbor		9.30×10^{-9}	9.30×10^{-9}	9.30×10^{-9}	9.30×10^{-9}	0.00	0.00	0.00	0.00	0.00
Portsmouth		2.30×10^{-7}	2.30×10^{-7}	2.30×10^{-7}	2.30×10^{-7}	0.00	0.00	0.00	0.00	0.00
Norfolk		2.10×10^{-5}	2.10×10^{-5}	2.10×10^{-5}	2.10×10^{-5}	0.00	0.00	0.00	0.00	0.00
Kesselring		4.10×10^{-12}	4.10×10^{-12}	4.10×10^{-12}	4.10×10^{-12}	0.00	0.00	0.00	0.00	0.00
	Total	2.24×10^{-5}	2.24×10^{-5}	8.74×10^{-5}	2.33×10^{-5}	8.50×10^{-7}	4.00×10^{-6}	1.80×10^{-5}	9.00×10^{-8}	5.00×10^{-5}
WATER POOL STORAGE AT ALL SITES*										
INEL		0.00	0.00	0.00	8.50×10^{-7}	8.50×10^{-7}	0.00	0.00	0.00	0.00
Hanford		0.00	0.00	0.00	0.00	0.00	4.00×10^{-6}	0.00	0.00	0.00
Savannah River		0.00	0.00	0.00	0.00	0.00	0.00	1.80×10^{-5}	0.00	0.00
Nevada Test Site		0.00	0.00	0.00	0.00	0.00	0.00	0.00	9.00×10^{-8}	0.00
Oak Ridge		0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	5.0×10^{-5}
Puget Sound		1.20×10^{-6}	6.50×10^{-5}	6.50×10^{-5}	6.50×10^{-5}	0.00	0.00	0.00	0.00	0.00
Pearl Harbor		9.30×10^{-9}	7.00×10^{-5}	7.00×10^{-5}	7.00×10^{-5}	0.00	0.00	0.00	0.00	0.00
Portsmouth		2.30×10^{-7}	2.30×10^{-5}	2.30×10^{-5}	2.30×10^{-5}	0.00	0.00	0.00	0.00	0.00
Norfolk		2.10×10^{-5}	1.40×10^{-4}	1.40×10^{-4}	1.40×10^{-4}	0.00	0.00	0.00	0.00	0.00
Kesselring		4.10×10^{-12}	4.10×10^{-5}	4.10×10^{-5}	4.10×10^{-5}	0.00	0.00	0.00	0.00	0.00
	Total	2.24×10^{-5}	3.39×10^{-4}	3.39×10^{-4}	3.40×10^{-4}	8.50×10^{-7}	4.00×10^{-6}	1.80×10^{-5}	9.00×10^{-8}	5.00×10^{-5}

*Under No Action alternative, dry storage at Naval Nuclear Propulsion Program sites

**Includes dry storage and water pool examination under this alternative

Table F-2. Number of fatal cancers from a maximum foreseeable accident (fatalities per accident over a 50-year period to general population within a 50-mile radius of site).

		No Action	Decentralization-No Examination	Decentralization-Puget Sound Exam	Decentralization-INEL Exam	Planning Basis/Regionalization/Centralization-INEL	Regionalization/Centralization-Hanford	Regionalization/Centralization-Savannah River	Regionalization/Centralization-Nevada Test Site	Regionalization/Centralization-Oak Ridge
DRY STORAGE AT NAVAL NUCLEAR PROPULSION PROGRAM SITES, WATER POOL STORAGE AT DOE SITES										
INEL		0.00	0.00	0.00	1.70×10^{-2}	1.70×10^{-2}	0.00	0.00	0.00	0.00
Hanford		0.00	0.00	0.00	0.00	0.00	4.70×10^{-2}	0.00	0.00	0.00
Savannah River		0.00	0.00	0.00	0.00	0.00	0.00	4.80	0.00	0.00
Nevada Test Site		0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.80×10^{-1}	0.00
Oak Ridge		0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	8.40
Puget Sound		1.7×10^{-2}	1.7×10^{-2}	5.1×10^{-1}**	1.7×10^{-2}	0.00	0.00	0.00	0.00	0.00
Pearl Harbor		2.60×10^{1}	2.60×10^{1}	2.60×10^{1}	2.60×10^{1}	0.00	0.00	0.00	0.00	0.00
Portsmouth		9.00	9.00	9.00	9.00	0.00	0.00	0.00	0.00	0.00
Norfolk		1.6×10^{1}	1.6×10^{1}	1.6×10^{1}	1.6×10^{1}	0.00	0.00	0.00	0.00	0.00
Kesselring		7.50	7.50	7.50	7.50	0.00	0.00	0.00	0.00	0.00
	Max	2.60×10^{1}	2.60×10^{1}	2.60×10^{1}	2.60×10^{1}	1.70×10^{-2}	4.70×10^{-2}	4.80	1.80×10^{-1}	8.40
WATER POOL STORAGE AT ALL SITES*										
INEL		0.00	0.00	0.00	1.70×10^{-2}	1.70×10^{-2}	0.00	0.00	0.00	0.00
Hanford		0.00	0.00	0.00	0.00	0.00	4.70×10^{-2}	0.00	0.00	0.00
Savannah River		0.00	0.00	0.00	0.00	0.00	0.00	4.80	0.00	0.00
Nevada Test Site		0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.80×10^{-1}	0.00
Oak Ridge		0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	8.40
Puget Sound		1.7×10^{-2}	5.1×10^{-1}	5.1×10^{-1}	5.1×10^{-1}	0.00	0.00	0.00	0.00	0.00
Pearl Harbor		2.60×10^{1}	1.10	1.10	1.10	0.00	0.00	0.00	0.00	0.00
Portsmouth		9.00	3.40×10^{-1}	3.40×10^{-1}	3.40×10^{-1}	0.00	0.00	0.00	0.00	0.00
Norfolk		1.6×10^{1}	6.0×10^{-1}	6.0×10^{-1}	6.0×10^{-1}	0.00	0.00	0.00	0.00	0.00
Kesselring		7.50	2.50×10^{-1}	2.50×10^{-1}	2.50×10^{-1}	0.00	0.00	0.00	0.00	0.00
	Max	2.60×10^{1}	1.10	1.10	1.10	1.70×10^{-2}	4.70×10^{-2}	4.80	1.80×10^{-1}	8.40

*Under No Action alternative, dry storage at Naval Nuclear Propulsion Program sites

**Includes dry storage and water pool examination under this alternative

190

Table F-3. Most severe risk from a facility accident (probability of fatalities per year per accident to general population within a 50-mile radius of site).

		No Action	Decentralization-No Examination	Decentralization-Puget Sound Exam	Decentralization-INEL Exam	Planning Basis/ Regionalization Centralization-INEL	Regionalization/ Centralization-Hanford	Regionalization/ Centralization-Savannah River	Regionalization/ Centralization-Nevada Test Site	Regionalization/ Centralization-Oak Ridge
DRY STORAGE AT NAVAL NUCLEAR PROPULSION PROGRAM SITES, WATER POOL STORAGE AT DOE SITES										
INEL		0.00	0.00	0.00	1.70×10^{-7}	1.70×10^{-7}	0.00	0.00	0.00	0.00
Hanford		0.00	0.00	0.00	0.00	0.00	4.70×10^{-7}	0.00	0.00	0.00
Savannah River		0.00	0.00	0.00	0.00	0.00	0.00	9.60×10^{-6}	0.00	0.00
Nevada Test Site		0.00	0.00	0.00	0.00	0.00	0.00	0.00	7.20×10^{-8}	0.00
Oak Ridge		0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	8.40×10^{-6}
Puget Sound		1.7×10^{-7}	1.7×10^{-7}	5.10×10^{-6}**	1.7×10^{-7}	0.00	0.00	0.00	0.00	0.00
Pearl Harbor		2.60×10^{-4}	2.60×10^{-4}	2.60×10^{-4}	2.60×10^{-4}	0.00	0.00	0.00	0.00	0.00
Portsmouth		9.00×10^{-7}	9.00×10^{-7}	9.00×10^{-7}	9.00×10^{-7}	0.00	0.00	0.00	0.00	0.00
Norfolk		1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	0.00	0.00	0.00	0.00	0.00
Kesselring		7.50×10^{-7}	7.50×10^{-7}	7.50×10^{-7}	7.50×10^{-7}	0.00	0.00	0.00	0.00	0.00
	Max	2.60×10^{-4}	2.60×10^{-4}	2.60×10^{-4}	2.60×10^{-4}	1.70×10^{-7}	4.70×10^{-7}	9.60×10^{-6}	7.2×10^{-8}	8.40×10^{-6}
WATER POOL STORAGE AT ALL SITES*										
INEL		0.00	0.00	0.00	1.70×10^{-7}	1.70×10^{-7}	0.00	0.00	0.00	0.00
Hanford		0.00	0.00	0.00	0.00	0.00	4.70×10^{-7}	0.00	0.00	0.00
Savannah River		0.00	0.00	0.00	0.00	0.00	0.00	9.60×10^{-6}	0.00	0.00
Nevada Test Site		0.00	0.00	0.00	0.00	0.00	0.00	0.00	7.20×10^{-8}	0.00
Oak Ridge		0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	8.40×10^{-6}
Puget Sound		1.7×10^{-7}	5.1×10^{-6}	5.1×10^{-6}	5.1×10^{-6}	0.00	0.00	0.00	0.00	0.00
Pearl Harbor		2.60×10^{-4}	1.10×10^{-5}	1.10×10^{-5}	1.10×10^{-5}	0.00	0.00	0.00	0.00	0.00
Portsmouth		9.00×10^{-7}	3.40×10^{-6}	3.40×10^{-6}	3.40×10^{-6}	0.00	0.00	0.00	0.00	0.00
Norfolk		1.6×10^{-5}	6.0×10^{-6}	6.0×10^{-6}	6.0×10^{-6}	0.00	0.00	0.00	0.00	0.00
Kesselring		7.50×10^{-7}	2.50×10^{-6}	2.50×10^{-6}	2.50×10^{-6}	0.00	0.00	0.00	0.00	0.00
	Max	2.60×10^{-4}	1.10×10^{-5}	1.10×10^{-5}	1.10×10^{-5}	1.70×10^{-7}	4.70×10^{-7}	9.60×10^{-6}	7.20×10^{-8}	8.40×10^{-6}

*Under No Action alternative, dry storage at Naval Nuclear Propulsion Program sites

**Includes dry storage and water pool examination under this alternative

191

Table F-4. Risk of fatal cancers by alternative (probability of fatalities per year per accident to general population within a 50-mile radius of site).

	No Action	Decentralization-No Examination	Decentralization-Puget Sound Exam	Decentralization-INEL Exam	Planning Basis/ Regionalization Centralization-INEL	Regionalization/ Centralization-Hanford	Regionalization/ Centralization-Savannah River	Regionalization/ Centralization-Nevada Test Site	Regionalization/ Centralization-Oak Ridge
Normal Operations Risk Dry Storage At Navy Sites, Water Pool Storage At DOE Sites	2.24×10^{-5}	2.24×10^{-5}	8.74×10^{-5}	2.33×10^{-5}	8.50×10^{-7}	4.00×10^{-6}	1.80×10^{-5}	9.00×10^{-8}	5.00×10^{-5}
Normal Operations Risk Water Pool Storage At All Sites	2.24×10^{-5}	3.39×10^{-4}	3.39×10^{-4}	3.40×10^{-4}	8.50×10^{-7}	4.00×10^{-6}	1.80×10^{-5}	9.00×10^{-8}	5.00×10^{-5}
Most Severe Risk From A Facility Accident Dry Storage At Naval Nuclear Propulsion Program Sites, Water Pool Storage At DOE Sites	2.60×10^{-4} (1)	2.60×10^{-4} (1)	2.60×10^{-4} (1)	2.60×10^{-4} (1)	1.70×10^{-7} (2)	4.70×10^{-7} (2)	9.60×10^{-6} (1)	7.20×10^{-8} (1)	8.40×10^{-6} (1)
Most Severe Risk From A Facility Accident Water Pool Storage At All Sites	2.60×10^{-4} (1)	1.10×10^{-5} (2)	1.10×10^{-5} (2)	1.10×10^{-5} (2)	1.70×10^{-7} (2)	4.70×10^{-7} (2)	9.60×10^{-6} (1)	7.2×10^{-8} (1)	8.40×10^{-6} (1)

(1) Accident initiator - Airplane crash
(2) Accident initiator - Drained water pool

192

Table F-5. Impacts from naval spent nuclear fuel facility radiological accidents for the No Action alternative.

Accident Description	Probability (per year)	Consequences to Public (fatalities per accident)	Risk to Public (fatalities)	Dose to Worker (rem)	Dose to MOI (rem)
DRY STORAGE ACCIDENTS					
Mechanical Damage					
Puget Sound	1.0×10^{-5}	1.7×10^{-2}	1.7×10^{-7}	5.6×10^{-2}	3.9×10^{-2}
Pearl Harbor	1.0×10^{-5}	3.0×10^{-2}	3.0×10^{-7}	5.6×10^{-2}	2.1×10^{-2}
Norfolk	1.0×10^{-5}	1.8×10^{-2}	1.8×10^{-7}	5.6×10^{-2}	8.1×10^{-2}
Portsmouth	1.0×10^{-5}	1.0×10^{-2}	1.0×10^{-7}	5.6×10^{-2}	4.2×10^{-2}
Kesselring	1.0×10^{-5}	7.4×10^{-3}	7.4×10^{-8}	5.6×10^{-2}	8.1×10^{-3}
Airplane Crash					
Pearl Harbor	1.0×10^{-5}	26	2.6×10^{-4}	92	19
Norfolk	1.0×10^{-6}	16	1.6×10^{-5}	92	72
Portsmouth	1.0×10^{-7}	9.0	9.0×10^{-7}	92	38
Kesselring	1.0×10^{-7}	7.5	7.5×10^{-7}	92	7.7

Table F-6. Impacts from naval spent nuclear fuel facility radiological accidents for Decentralization alternatives.

Accident Description	Probability (per year)	Consequences to Public (fatalities per accident)	Risk to Public (fatalities)	Dose to Worker (rem)	Dose to MOI (rem)
WET STORAGE AND EXAMINATION ACCIDENTS					
*Information applicable only for full examinations at INEL.					
Drained Water Pool					
*INEL	1.0×10^{-5}	1.7×10^{-2}	1.7×10^{-7}	2.1	1.7×10^{-2}
Puget Sound	1.0×10^{-5}	5.1×10^{-1}	5.1×10^{-6}	2.1	1.4
Pearl Harbor	1.0×10^{-5}	1.1	1.1×10^{-5}	2.1	7.9×10^{-1}
Norfolk	1.0×10^{-5}	6.0×10^{-1}	6.0×10^{-6}	2.1	3.0
Portsmouth	1.0×10^{-5}	3.4×10^{-1}	3.4×10^{-6}	2.1	1.6
Kesselring	1.0×10^{-5}	2.5×10^{-1}	2.5×10^{-6}	2.1	2.9×10^{-1}
Accidental Criticality					
*INEL	1.0×10^{-5}	6.4×10^{-3}	6.4×10^{-8}	8.0	9.2×10^{-3}
Puget Sound	1.0×10^{-5}	2.8×10^{-1}	2.8×10^{-6}	8.0	1.3
Pearl Harbor	1.0×10^{-5}	6.0×10^{-1}	6.0×10^{-6}	8.0	6.7×10^{-1}
Norfolk	1.0×10^{-5}	3.5×10^{-1}	3.5×10^{-6}	8.0	2.7
Portsmouth	1.0×10^{-5}	1.5×10^{-1}	1.5×10^{-6}	8.0	1.4
Kesselring	1.0×10^{-5}	1.1×10^{-1}	1.1×10^{-6}	8.0	2.3×10^{-1}
Mechanical Damage					
*INEL	1.0×10^{-5}	5.3×10^{-6}	5.3×10^{-11}	5.2×10^{-4}	2.6×10^{-6}
Puget Sound	1.0×10^{-5}	7.2×10^{-5}	7.2×10^{-10}	5.2×10^{-4}	1.7×10^{-4}
Pearl Harbor	1.0×10^{-5}	1.5×10^{-4}	1.5×10^{-9}	5.2×10^{-4}	9.3×10^{-5}
Norfolk	1.0×10^{-5}	8.0×10^{-5}	8.0×10^{-10}	5.2×10^{-4}	3.5×10^{-4}
Portsmouth	1.0×10^{-5}	5.6×10^{-5}	5.6×10^{-10}	5.2×10^{-4}	1.9×10^{-4}
Kesselring	1.0×10^{-5}	6.0×10^{-5}	6.0×10^{-10}	5.2×10^{-4}	3.6×10^{-5}
Airplane Crash					
Pearl Harbor	2.0×10^{-5}	4.6×10^{-2}	9.2×10^{-7}	1.6×10^{-1}	2.8×10^{-2}
Norfolk	4.0×10^{-7}	2.4×10^{-2}	9.6×10^{-9}	1.6×10^{-1}	1.1×10^{-1}
Kesselring	2.0×10^{-7}	1.8×10^{-2}	3.6×10^{-9}	1.6×10^{-1}	1.1×10^{-2}
HEPA Filter Fire					
*INEL	5.0×10^{-4}	5.3×10^{-5}	2.7×10^{-8}	2.4×10^{-3}	2.5×10^{-5}
Puget Sound	5.0×10^{-4}	6.4×10^{-4}	3.2×10^{-7}	2.4×10^{-3}	1.6×10^{-3}
Pearl Harbor	5.0×10^{-4}	1.2×10^{-3}	6.0×10^{-7}	2.4×10^{-3}	8.7×10^{-4}

Table F-6. Impacts from naval spent nuclear fuel facility radiological accidents for Decentralization alternatives. (Cont)

Accident Description	Probability (per year)	Consequences to Public (fatalities per accident)	Risk to Public (fatalities)	Dose to Worker (rem)	Dose to MOI (rem)
Norfolk	5.0×10^{-4}	6.9×10^{-4}	3.5×10^{-7}	2.4×10^{-3}	3.3×10^{-3}
WET STORAGE AND EXAMINATION ACCIDENTS					
*Information applicable only for full examinations at INEL.					
Portsmouth	5.0×10^{-4}	3.9×10^{-4}	2.0×10^{-7}	2.4×10^{-3}	1.7×10^{-3}
Kesselring	5.0×10^{-4}	3.3×10^{-4}	1.7×10^{-7}	2.4×10^{-3}	3.5×10^{-4}
Minor Water Pool Leak					
*INEL	1.0×10^{-1}	1.3×10^{-8}	1.3×10^{-9}	N/A	2.5×10^{-9}
Puget Sound	1.0×10^{-1}	4.2×10^{-9}	4.2×10^{-10}	N/A	3.2×10^{-10}
Pearl Harbor	1.0×10^{-1}	4.6×10^{-10}	4.6×10^{-11}	N/A	1.3×10^{-10}
Norfolk	1.0×10^{-1}	1.8×10^{-9}	1.8×10^{-10}	N/A	2.7×10^{-10}
Portsmouth	1.0×10^{-1}	1.4×10^{-9}	1.4×10^{-10}	N/A	1.3×10^{-10}
Kesselring	1.0×10^{-1}	8.5×10^{-9}	8.5×10^{-10}	N/A	6.0×10^{-9}
DRY STORAGE ACCIDENTS					
Mechanical Damage					
Puget Sound	1.0×10^{-5}	1.7×10^{-2}	1.7×10^{-7}	5.6×10^{-2}	3.9×10^{-2}
Pearl Harbor	1.0×10^{-5}	3.0×10^{-2}	3.0×10^{-7}	5.6×10^{-2}	2.1×10^{-2}
Norfolk	1.0×10^{-5}	1.8×10^{-2}	1.8×10^{-7}	5.6×10^{-2}	8.1×10^{-2}
Portsmouth	1.0×10^{-5}	1.0×10^{-2}	1.0×10^{-7}	5.6×10^{-2}	4.2×10^{-2}
Kesselring	1.0×10^{-5}	7.4×10^{-3}	7.4×10^{-8}	5.6×10^{-2}	8.1×10^{-3}
Airplane Crash					
Pearl Harbor	1.0×10^{-5}	26	2.6×10^{-4}	92	19
Norfolk	1.0×10^{-6}	16	1.6×10^{-5}	92	72
Portsmouth	1.0×10^{-7}	9.0	9.0×10^{-7}	92	38
Kesselring	1.0×10^{-7}	7.5	7.5×10^{-7}	92	7.7
DRY CELL ACCIDENTS					
Mechanical Damage					
*INEL	1.0×10^{-4}	3.5×10^{-4}	3.5×10^{-8}	1.0×10^{-1}	2.2×10^{-4}
Loss of Shielding					
*INEL	1.0×10^{-5}	3.0×10^{-19}	3.0×10^{-24}	7.2×10^{-5}	9.3×10^{-17}

Table F-7. Impacts from naval spent nuclear fuel facility radiological accidents for Planning Basis, Centralization at INEL, and Regionalization at INEL alternatives.

Accident Description	Probability (per year)	Consequences to Public (fatalities per accident)	Risk to Public (fatalities)	Dose to Worker (rem)	Dose to MOI (rem)
WET STORAGE AND EXAMINATION ACCIDENTS					
Drained Water Pool					
INEL	1.0×10^{-5}	1.7×10^{-2}	1.7×10^{-7}	2.1	1.7×10^{-2}
Accidental Criticality					
INEL	1.0×10^{-5}	6.4×10^{-3}	6.4×10^{-8}	8.0	9.2×10^{-3}
Mechanical Damage					
INEL	1.0×10^{-5}	5.3×10^{-6}	5.3×10^{-11}	5.2×10^{-4}	2.6×10^{-6}
HEPA Filter Fire					
INEL	5.0×10^{-4}	5.3×10^{-5}	2.7×10^{-8}	2.4×10^{-3}	2.5×10^{-5}
Minor Water Pool Leak					
INEL	1.0×10^{-1}	1.3×10^{-8}	1.3×10^{-9}	N/A	2.5×10^{-9}
DRY STORAGE ACCIDENTS					
Mechanical Damage					
INEL	1.0×10^{-5}	4.9×10^{-4}	4.9×10^{-9}	5.6×10^{-2}	4.6×10^{-4}
DRY CELL ACCIDENTS					
Mechanical Damage					
INEL	1.0×10^{-4}	3.5×10^{-4}	3.5×10^{-8}	1.0×10^{-1}	2.2×10^{-4}
Loss of Shielding					
INEL	1.0×10^{-5}	3.0×10^{-19}	3.0×10^{-24}	7.2×10^{-5}	9.3×10^{-17}

Table F-8. Impacts from naval spent nuclear fuel facility radiological accidents for Regionalization or Centralization at other DOE sites alternatives.

Information applicable only to DOE site selected for Regionalization or Centralization.

Accident Description	Probability (per year)	Consequences to Public (fatalities per accident)	Risk to Public (fatalities)	Dose to Worker (rem)	Dose to MOI (rem)
WET STORAGE AND EXAMINATION ACCIDENTS					
Drained Water Pool					
Savannah River	1.0×10^{-5}	1.1×10^{-1}	1.1×10^{-6}	2.1	1.6×10^{-2}
Hanford	1.0×10^{-5}	4.7×10^{-2}	4.7×10^{-7}	2.1	6.3×10^{-3}
Nevada Test Site	1.0×10^{-5}	1.9×10^{-3}	1.9×10^{-8}	2.1	3.3×10^{-2}
Oak Ridge	1.0×10^{-5}	1.8×10^{-1}	1.8×10^{-6}	2.1	5.2
Accidental Criticality					
Savannah River	1.0×10^{-5}	4.5×10^{-2}	4.5×10^{-7}	8.0	9.4×10^{-3}
Hanford	1.0×10^{-5}	1.6×10^{-2}	1.6×10^{-7}	8.0	2.8×10^{-3}
Nevada Test Site	1.0×10^{-5}	7.0×10^{-4}	7.0×10^{-9}	8.0	2.0×10^{-2}
Oak Ridge	1.0×10^{-5}	8.8×10^{-2}	8.8×10^{-7}	8.0	4.7
Mechanical Damage					
Savannah River	1.0×10^{-5}	2.0×10^{-5}	2.0×10^{-10}	5.2×10^{-4}	2.2×10^{-6}
Hanford	1.0×10^{-5}	8.6×10^{-6}	8.6×10^{-11}	5.2×10^{-4}	9.8×10^{-7}
Nevada Test Site	1.0×10^{-5}	5.6×10^{-7}	5.6×10^{-12}	5.2×10^{-4}	4.6×10^{-6}
Oak Ridge	1.0×10^{-5}	3.4×10^{-5}	3.4×10^{-10}	5.2×10^{-4}	5.9×10^{-4}
Airplane Crash					
Savannah River	2.0×10^{-6}	6.1×10^{-3}	1.2×10^{-8}	1.6×10^{-1}	6.4×10^{-4}
Oak Ridge	1.0×10^{-6}	1.0×10^{-2}	1.0×10^{-8}	1.6×10^{-1}	1.8×10^{-1}
Nevada Test Site	4.0×10^{-7}	1.7×10^{-4}	6.8×10^{-11}	1.6×10^{-1}	1.3×10^{-3}
HEPA Filter Fire					
Savannah River	5.0×10^{-4}	1.3×10^{-4}	6.5×10^{-8}	2.4×10^{-3}	2.1×10^{-5}
Hanford	5.0×10^{-4}	5.3×10^{-5}	2.7×10^{-8}	2.4×10^{-3}	7.0×10^{-6}
Nevada Test Site	5.0×10^{-4}	5.7×10^{-6}	2.9×10^{-9}	2.4×10^{-3}	4.3×10^{-5}
Oak Ridge	5.0×10^{-4}	2.2×10^{-4}	1.1×10^{-7}	2.4×10^{-3}	5.7×10^{-3}
Minor Water Leak					
Savannah River	1.0×10^{-1}	1.3×10^{-9}	1.3×10^{-10}	N/A	7.9×10^{-10}
Hanford	1.0×10^{-1}	1.7×10^{-10}	1.7×10^{-11}	N/A	9.9×10^{-12}
Nevada Test Site	1.0×10^{-1}	1.4×10^{-9}	1.4×10^{-10}	N/A	2.5×10^{-9}

Table F-8. Impacts from naval spent nuclear fuel facility radiological accidents for Regionalization or Centralization at other DOE sites alternatives. (Cont)

Information applicable only to DOE sites selected for Regionalization or Centralization.

Accident Description	Probability (per year)	Consequences to Public (fatalities per accident)	Risk to Public (fatalities)	Dose to Worker (rem)	Dose to MOI (rem)
Oak Ridge	1.0×10^{-1}	3.9×10^{-9}	3.9×10^{-10}	N/A	1.5×10^{-9}
DRY STORAGE ACCIDENTS					
Mechanical Damage					
Savannah River	1.0×10^{-5}	3.0×10^{-3}	3.0×10^{-8}	5.6×10^{-2}	4.9×10^{-4}
Hanford	1.0×10^{-5}	1.3×10^{-3}	1.3×10^{-8}	5.6×10^{-2}	1.7×10^{-4}
Nevada Test Site	1.0×10^{-5}	5.3×10^{-5}	5.3×10^{-10}	5.6×10^{-2}	8.8×10^{-4}
Oak Ridge	1.0×10^{-5}	5.1×10^{-3}	5.1×10^{-8}	5.6×10^{-2}	1.4×10^{-1}
Airplane Crash					
Savannah River	3.0×10^{-7}	2.8	8.4×10^{-7}	92	4.7×10^{-1}
Oak Ridge	3.0×10^{-7}	4.7	1.4×10^{-6}	92	120
DRY CELL ACCIDENTS					
Mechanical Damage					
Savannah River	1.0×10^{-4}	1.4×10^{-3}	1.4×10^{-7}	1.0×10^{-1}	2.4×10^{-4}
Hanford	1.0×10^{-4}	5.3×10^{-4}	5.3×10^{-8}	1.0×10^{-1}	7.1×10^{-5}
Nevada Test Site	1.0×10^{-4}	3.7×10^{-5}	3.7×10^{-9}	1.0×10^{-1}	4.0×10^{-4}
Oak Ridge	1.0×10^{-4}	2.5×10^{-3}	2.5×10^{-7}	1.0×10^{-1}	5.8×10^{-2}
Loss of Shielding					
Savannah River	1.0×10^{-5}	3.0×10^{-16}	3.0×10^{-21}	7.2×10^{-5}	6.7×10^{-15}
Hanford	1.0×10^{-5}	4.9×10^{-24}	4.9×10^{-29}	7.2×10^{-5}	3.3×10^{-23}
Nevada Test Site	1.0×10^{-5}	3.7×10^{-37}	3.7×10^{-42}	7.2×10^{-5}	6.3×10^{-11}
Oak Ridge	1.0×10^{-5}	7.5×10^{-6}	7.5×10^{-11}	7.2×10^{-5}	1.2×10^{-2}
Airplane Crash					
Savannah River	2.0×10^{-6}	4.8	9.6×10^{-6}	160	8.2×10^{-1}
Oak Ridge	1.0×10^{-6}	8.4	8.4×10^{-6}	160	350
Nevada Test Site	4.0×10^{-7}	1.8×10^{-1}	7.2×10^{-8}	160	1.6

concentrations were then compared against Emergency Release Planning Guide (ERPG) levels as a means of evaluating their effects. ERPG values are specific for each substance and provide an estimate of the airborne concentration thresholds above which one can reasonably observe adverse effects. Exposure to an ERPG-1 level could result in a very mild effect whereas exposure to an ERPG-3 level could result in a life-threatening health effect. For the postulated accident involving a chemical spill and fire, on-site personnel (worker) could be exposed to concentrations of hydrochloric acid, phosgene, sulfuric acid, and sodium hydroxide above ERPG-3 levels which indicates a potential for long-term health effects. However, no member of the general public located off-site would be expected to be exposed to levels above ERPG-3 except for Oak Ridge where sulfuric acid and sodium hydroxide concentrations could exceed ERPG-3. For the postulated accident involving a diesel fuel fire, on-site personnel could be exposed to concentrations of sulfur dioxide and oxides of nitrogen above ERPG-3 levels. No member of the general public located off-site would be expected to be exposed to levels above ERPG-3 except for Oak Ridge where sulfur dioxide and oxides of nitrogen concentrations could exceed ERPG-3 and one shipyard location (Norfolk) where nitric oxide concentrations could exceed ERPG-3 under severe meteorological conditions. However, for both postulated accidents, the accident analyses did not include evacuation of on-site or off-site personnel and it is expected that chemical exposures would be below ERPG-3 levels because actions such as evacuation would be used to reduce the effects on the public and workers.

Fugitive Dust Analysis

The FDM computer code was used to estimate the fugitive dust concentrations that could result from the construction of a water pool facility at the alternate locations. It was determined that the release of fugitive dust would not result in any adverse effects for any of the alternate locations.

Other Impacts

The radiological impact of accidents on the environs of a facility was determined by examining the area that could be contaminated following such an event. Calculations using average meteorological conditions were performed for each accident scenario. These calculations determined the extent of the contamination which causes only a small increase in background radiation from naturally occurring sources. For most facilities and most accidents, the contaminated area was confined to the boundaries of the site. For a few cases, the casualty scenarios did result in contaminated land outside the site boundaries; however, the total land contaminated for those scenarios (inside and outside the boundary) was no more than 207 acres. The impact of this contamination would be temporary while the area was isolated and remediation efforts completed.

F.1 RADIOLOGICAL ISSUES FROM NAVAL SPENT NUCLEAR FUEL INSPECTIONS AND STORAGE

Naval spent nuclear fuel is currently examined and stored at the Naval Reactors Facility's Expended Core Facility (ECF) at the DOE Idaho National Engineering Laboratory (INEL). The INEL-ECF is a large laboratory facility used to receive, examine, and ship naval spent nuclear fuel and irradiated test specimen assemblies. Enclosed work areas at INEL-ECF include an array of interconnected reinforced concrete water pools which permit visual observation of naval spent nuclear fuel during handling and inspection while shielding workers from radiation. Adjacent to the water pools are shielded cells used for operations which must be performed dry. One of the water pools contains transfer canals that will link the water pools with a proposed Dry Cell Project, which would provide a location for preparation of spent fuel in a dry, enclosed environment.

The proposed Dry Cell Facility will consist of a shielded, radiologically controlled area built of structural steel and concrete with remotely operated equipment necessary to examine fuel modules.

The Organization for Economic Co-operation and Development (OECD) of the Nuclear Energy Agency (NEA) reported that extensive safety analysis has shown that pool storage of Zircaloy-clad fuel is a very safe option which can last for decades (NEA 1993). The external hazards, such as earthquakes and aircraft crashes, are potential threats for these facilities (loss of coolant) but appropriate siting, design, and additional shielding can cope with these hazards. Dry storage has not yet generally been carried out on a very large scale but it is anticipated that long-term storage in adequate canisters is a very safe practice even against earthquakes and aircraft crashes.

Several technologies are being used currently for the storage of spent fuel at reactor sites and at sites away from reactors. Both wet (pool) storage facilities and dry storage facilities (buildings and containers) are used on a commercial scale.

The safety of spent fuel storage has been extensively evaluated. The U.S. Nuclear Regulatory Commission (NRC) reported in the "Waste Confidence Decision" of 1984 that there is reasonable assurance that spent fuel can be stored safely and without significant environmental impact in reactor pools or in spent fuel storage installations (NUREG 1984). For both dry storage and wet storage, the NRC stated its belief that current storage technologies are capable of providing safe storage for at

least 30 years beyond the active lifetime of the reactor facility. The NRC also concluded that the possibility of a major accident or sabotage at a spent fuel storage facility with radiological consequences for the public is extremely remote.

Considerable experience has been gained in the transport of spent fuel elements and in the consequent safety-related development of suitable transportation casks. This experience has made it possible to develop a concept for dry storage of spent fuel elements within transportation casks; dry storage containers generally have not been the transportation casks themselves.

The concept of a cask which could be used for both transportation and storage has been licensed in the United States in the framework of a policy of dry storage in Independent Spent Fuel Storage Installations (CFR 1993). According to this policy, the reactor operators are entitled to store the spent fuel elements, which have cooled in a pool for at least one year after discharge from the reactor, in specially licensed containers under dry conditions for 20 years or more. A number of storage casks have received official approval for that purpose.

F.1.1 Normal Operations

Current practice for examination of naval spent nuclear fuel at ECF includes removal of upper and lower non-fuel bearing structures, visual examination, measurement of key dimensions, collection of specimens, and loading into a shipping cask. Temporary storage of spent fuel at INEL-ECF is required since fuel is, at times, received into the facility faster than it can be examined and shipped out of the facility. In addition, a small amount of spent fuel is selected for retention as library specimens for future reference and examination. Routine releases to the atmosphere were evaluated at all locations based on measured releases from INEL-ECF. Each location was evaluated using releases equivalent to those of INEL-ECF. Each location's specific population and meteorology were then used to produce estimated consequences.

F.1.1.1 Water Pool Storage. Wet storage is a highly developed technique and it is the standard method used worldwide for storage of spent fuel. While in wet storage pools, temperatures, pressures, and radiation fluxes are lower than in the reactor, so there is no intrinsic driving force for the sudden release of a major fraction of the radioactive materials contained in the stored spent fuel.

The Zircaloy cladding of naval spent nuclear fuel is an efficient barrier against fission product release during handling and storage of spent fuel. Given adequate control of water purity, Zircaloy resists corrosion in water during the long-term storage conditions of fuel assemblies. At the end of its service life, the fuel is covered with a tightly adhering oxide layer formed at high temperatures which is a major factor that inhibits further corrosion during storage.

Direct exposure to radiation of persons working in storage facilities can occur during such activities as handling of fuel casks and fuel assemblies, handling of contaminated filters, and repair and maintenance work. Experience shows that, in common with other fuel cycle facilities, the risk of increased occupational exposure arises when any maintenance or unusual operations are carried out. Such increased exposures can, however, generally be minimized by good planning, adequate redundancy of critical components, paying particular attention to the design of those items that are liable to become contaminated from the point of view of repair and maintenance, and by the use of local shielding and equipment decontamination procedures. Systems and components that are important in this context include:

- pool water cooling and makeup systems;
- filter equipment for purification of pool water;
- ventilation systems;
- equipment for temperature, water level, and leakage measurement in the fuel pools;
- hoists and handling systems for fuel assemblies; and
- equipment for handling and storage of other wastes.

Shielding from radiation is normally assured by providing a minimum depth of water above the fuel elements in storage to reduce the exposure rates. Fuel transfer mechanisms have limit switches and mechanical stops to prevent the inadvertent raising of fuel to the water surface. A high-integrity pool structure is needed in order to guarantee adequate containment of the pool water, but a limited loss of water resulting in a substantial reduction of the shielding layer is unlikely to involve high risks of exposures to personnel above operational limits since adequate countermeasures can be taken in time.

Storage of naval spent nuclear fuel in water pools is an alternative being evaluated at all DOE and Navy shipyard locations discussed above. Source terms for all locations were based on actual

releases reported by INEL-ECF in the past. Exposures due to downwind dispersion, water release, and direct radiation were calculated.

F.1.1.2 Dry Storage. Many thousands of spent fuel assemblies of different types have been stored for periods of time ranging from a couple of years to over 30 years in more than 20 different dry storage facilities. In general, the spent fuel behavior during storage has been excellent and no detrimental effects of dry storage on the integrity of the spent fuel have been detected (NEA 1993).

The dry storage of spent fuel is being used to a limited extent in several countries. In the United States, fuel was stored in dry wells at the INEL. Dry wells were used for the storage of a small amount of fuel at the Nevada Test Site as part of a large dry storage demonstration program. Storage started at the Climax deep dry wells (600 meters below the surface in granite) in 1979. In 1983, one fuel assembly underwent extensive non-destructive and destructive characterization. No problems requiring process changes were identified (NEA 1993).

Designs of metal casks for use in spent fuel storage have been in existence since the late 1970s. The casks are generally equipped with a double-lid system to ensure safe containment of contents. These casks have been subjected to a variety of tests and demonstrations since the early 1980s using both intact and consolidated fuel.

The DOE sponsored the demonstration of the storage of fuel in metal casks at the Morris storage facility in 1984 and 1985. The DOE entered into a cooperative agreement with Virginia Power, a United States' utility, to demonstrate the use of three types of metal casks. The Virginia Power Surry Nuclear Power Station has been licensed by the NRC for storage of spent fuel in metal casks.

Results of demonstration activities have shown the following (NEA 1993):

- radiation and thermal levels resulting from metal cask storage have been acceptable;
- no fuel failure has occurred during demonstration storage;
- no secondary wastes have arisen from the storage operation.

Storage of naval spent nuclear fuel in storage or shipping containers is an alternative being evaluated at all locations. Since no airborne releases are expected from routine dry storage activity, only the biological effects of direct radiation exposure to the on-site personnel and the public were determined.

F.1.1.3 Dry Cell Operations. The handling of naval spent nuclear fuel for research and development purposes in dry cells like the proposed Dry Cell Project was evaluated at selected DOE locations. The health effects due to routine airborne releases and direct radiation exposure were estimated.

F.1.2 Screening/Selection of Accidents for Detailed Examination

Accidents were considered for inclusion in detailed analyses if they were expected to contribute substantially to risk (defined as the product of the probability of occurrence of the accident times the consequence of the accident). Accidents were categorized into three types as either Abnormal Events, Design Basis Accidents, or Beyond Design Basis Accidents. These categories are characterized by their probability of occurrence as described further in Section F.1.3.7. Construction and industrial accidents are included in these categories.

In selecting accidents to include in detailed analyses, several considerations were utilized. Initiating events were reviewed including natural phenomena (earthquakes, volcanic activity, tornadoes, hurricanes and other natural events) and human initiated events (human error, equipment failures, fires, explosions, plane crashes, transportation accidents, and terrorism). Guiding principles were established, such as: the radioactive materials involved must be available in a dispersible form; there must be a mechanism available for release of such materials from the facility; and, there must be a mechanism available for off-site dispersion of the released materials. The pathways whereby members of the public can be affected from the nuclear aspects of spent fuel operations are direct exposure to radiation, inhalation of radioactive materials, or ingestion of radioactive materials. Recognizing these fundamental processes and pathways, accidents involving the following basic phenomena were identified:

- loss of shielding of radioactive materials,
- release of radioactive products to the environment due to overheating of fuel,
- release of radioactive products to the environment due to mechanical shock or damage or inadvertent breaching of fuel cladding or containment,

- an unplanned criticality,
- transportation accidents.

After the basic phenomena were identified, other references were consulted to ensure that all important accidents were considered. These included safety analysis reports, court decisions, other environmental impact statements, and summary documents such as the "Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Reactor Power Reactor Fuel" (NUREG 1979a) and "The Safety of the Nuclear Fuel Cycle" (NEA 1993).

Examining the kinds of accidents which could result in release of radioactive material to the environment or an increase in radiation levels shows that they can only occur if an accident produces severe conditions. Some types of accidents, such as procedure violations, spills of small volumes of water containing radioactive particles, or most other types of common human error, may occur more frequently than the more severe accidents analyzed. However, they do not involve enough radioactive material or radiation to result in a significant release to the environment or a meaningful increase in radiation levels. Stated another way, the very low consequences associated with these events produce smaller risks than those for the accidents analyzed, even when combined with a higher probability of occurrence. Consequently, they have not been included in the results presented in this Environmental Impact Statement.

Acts of terrorism are expected to result in consequences which are bounded by the results of accidents which were evaluated. Naval spent nuclear fuel is not considered to be attractive to terrorists due to the bulk of the fuel and containers and due to the high radiation fields involved with unshielded spent nuclear fuel. However, terrorist attacks on naval spent nuclear fuel during shipment were evaluated. The massive structure of the shipping containers used for naval spent nuclear fuel makes them an unlikely target of a terrorist attack. No such attacks have occurred in the nearly 40 years of rail shipments which have now travelled about 2 million kilometers. Thus, the probability of a terrorist attack on a shipment is judged to be no more than the probability of a rail accident which is listed in Section A.7.1.2.1 of Attachment A to Appendix D of this Environmental Impact Statement. The consequences of a terrorist attack are also judged to be no more severe than those listed for transportation accidents. Therefore, the same conclusions reached for transportation accidents apply to the risk to the extremely rugged shipping containers from terrorist attack during a shipment. In addition, during shipment, all naval spent nuclear fuel containers are accompanied by escorts who remain in contact with headquarters. In the event of an emergency, state and federal resources would be quickly summoned to stabilize the situation.

For an act of war, sabotage, or terrorist attack, it is likely the risk would be lower than calculated for the airplane crash because it should be less probable that a force would exist to disperse radioactive products into the atmosphere from a weapon as compared to the motive force of the fire assumed in the case of an airplane crash. For example, attacks on containers using anti-tank weapons would be less severe than the accidents analyzed because: (a) anti-tank weapons would cause a self-sealing penetration in the metal of a container, unlike that which is assumed from the airplane crash (impact from a 50-inch diameter engine rotor); (b) there is no explosive material inside the container, so it will not "blow up" as a tank would if hit by such a weapon (in a tank attack, the tank shells inside the turret detonate); (c) there would be no fire to disperse the radioactivity that is released when the container is breached, unlike an aircraft crash where the jet fuel will burn creating such a fire. The rugged design of containers and the thick walls of water pools, combined with the shock-absorbing nature of water with a free surface, reduce the effects of other types of explosive charges. It is not credible that a terrorist attack would result in a criticality or meltdown of spent nuclear fuel; however, in Section F.1.4.2.1.2, the consequences of a hypothetical criticality accident are presented. The risks associated with an accidental criticality are less than those associated with a drained water pool or an airplane crash into dry storage containers.

The effect of a terrorist attack or an act of sabotage is expected to be conservatively bounded by the limiting accident discussed at each facility under each alternative. For example, the most limiting accident involving naval spent nuclear fuel is described in this attachment to be an airplane crash into a shipping container at the Pearl Harbor Naval Shipyard. This accident would lead to 26 latent fatal cancers over the next 50 years in the population within 50 miles of the shipyard. Since the probability of the event is one chance in 100,000 per year, the risk would be 0.00026 latent fatal cancer fatalities per year or, in other words, about one chance in 4,000 of a single latent fatal cancer fatality over a year. This risk is shared among the approximately 820,000 people residing within 50 miles of the shipyard who would be expected to have over 2,000 cancer fatalities from all causes every year. For an act of war, sabotage, or terrorist attack, it is likely the risk would be lower than calculated because it should be less probable that a force would exist to disperse radioactive products into the atmosphere from a weapon as compared to the motive force of the fire assumed in the case of an airplane crash.

Accidents initiated at nearby facilities, by other activities unrelated to spent nuclear fuel handling or storage, or during construction of an ECF or dry cell type of facility, would not produce effects more severe than the sequences of events described. This is because naval spent nuclear fuel undergoing examination or in storage under the conditions of the alternatives evaluated would not need special conditions or uninterrupted operator attention to prevent overheating, failure of containment, or loss of shielding. Therefore, evacuation in response to an accident at some other facility would not compromise safety. This inherent safety, combined with the distance between naval spent nuclear fuel facilities and any other activities which might suffer a catastrophic accident, means that the accidents analyzed in this document produce conditions at a naval spent nuclear fuel facility which would be more severe than those for any hypothetical synergistic combination of events resulting from accidents at other, unrelated facilities. Therefore, such analyses have not been included in this evaluation.

The existence of common cause accidents at a facility has been considered. In general, only one spent nuclear fuel facility is located at a particular Navy site. However, it is possible for natural phenomena, like an earthquake, to produce more than one accident at some sites causing a situation resulting in the release of radioactive material into the atmosphere or an increase in radiation levels due to loss of shielding. However, the probability of two or more accidents having maximum consequences occur concurrently is less than the probability of the individual events. For example, if an earthquake affected the Naval Reactors Facility at INEL, a crane might fail causing damage to stored spent fuel, the water pool might drain, and shielding for the Dry Cell might be damaged. The impacts for this could conservatively be estimated by summing the consequences. A combined total of 2.8×10^{-2} fatal cancers are estimated. Similarly, consequences from spent nuclear fuel facilities within a DOE site could be combined to conservatively estimate site wide impacts. But again, the probability of a common cause event resulting in this number of consequences is lower than the probability of the individual accidents because the severity of impact will vary between facilities due to separation distances.

Several accident scenarios were developed for the handling and storage of naval spent nuclear fuel. All potential accidents were not evaluated, but cases which are considered to be more severe than all other reasonable accidents were analyzed. Each of these accident scenarios was evaluated at several locations using identical source terms. Like the evaluations for normal operations, population and meteorology data specific to each site were used to estimate site specific health effects.

F.1.2.1 Water Pool Storage. Six hypothetical accident scenarios were evaluated for naval spent nuclear fuel stored in water pools. These hypothetical sequences of events include a drainage of the water pool caused by an earthquake, an accidental criticality, mechanical damage due to operator error or crane failure, an airplane crash into the water pool facility, a fire in a high efficiency particulate air (HEPA) filter, and minor water pool leakage. Radiation exposure to on-site individuals, an individual at the site boundary, and the general population was estimated for airborne releases of radioactivity, water releases, and direct radiation exposure.

F.1.2.2 Dry Storage. Two hypothetical accident scenarios were evaluated for naval spent nuclear fuel stored in shipping containers. The first scenario postulates that a wind-driven missile crashes into storage casks, with mechanical damage causing a release of corrosion products into the environment. The second hypothetical scenario is based on an airplane crash into the dry storage area. Once again, radiation exposure to on-site individuals, an individual at the site boundary, and the general population was estimated for airborne releases, water releases, and direct radiation exposure.

F.1.2.3 Dry Cell Operations. Three hypothetical accidents were evaluated for naval spent nuclear fuel handled in dry cells at several locations. These scenarios include cutting into the fuel region or mechanical damage during examination work, partial loss of concrete shielding due to an earthquake, and an airplane crash into the dry cell facility. Once again, radiation exposure to on-site individuals, an individual at the site boundary, and the general population was estimated for airborne releases, water releases, and direct radiation exposure.

F.1.2.4 Shipboard Fire Involving Shipping Containers. Attachment A describes the historical practice of shipping naval spent nuclear fuel from Pearl Harbor Naval Shipyard to Puget Sound Naval Shipyard by ship where the containers are then transported to ECF by rail. Since 1962, there have been 17 shipments containing a total of 20 shipping containers. Even though there have not been any accidents involving these shipments, hypothetical accidents were evaluated near the Pearl Harbor and Puget Sound shipyards. The scenario involves a collision of the spent nuclear fuel ship with another ship which results in a fire. The radiation exposure to nearby individuals and the general population was estimated for airborne and water releases.

F.1.3 Analysis Methods for Evaluation of Radiation Exposure

F.1.3.1 General. An evaluation of normal operations and hypothetical accidents at the existing and proposed sites was performed to assess the possible radiation exposure to individuals due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential locations and the same accidents at any of the sites evaluated. With this approach,

it is possible to compare the incremental effect of the proposed alternative actions or the different impacts of the postulated accidents at the different sites. These locations include four naval shipyards (Portsmouth, Norfolk, Puget Sound, and Pearl Harbor), five Department of Energy facilities (INEL, Savannah River, Hanford, Nevada Test Site, and Oak Ridge), and the Kesselring Site.

F.1.3.2 Exposures to be Calculated. Radiation exposure to the following different individuals and the general population is calculated for normal operation of the spent fuel facility and for accident conditions:

- Worker (Worker). An individual located 100 meters (330 feet) from the radioactive material release point. (The impact of accidents on close-in workers is not calculated numerically but is discussed qualitatively for each accident in Section F.1.4.3 of this attachment.)

- Maximally exposed collocated worker (MCW). At DOE locations, a theoretical individual located at whichever is the greater of 0.4 mile from the facility area boundary or 75% of the distance to the nearest independent facility area. The MCW is not evaluated if the site boundary is closer than the MCW location. Thus, at shipyard locations and the Kesselring Site, the MCW is not specifically evaluated.

- Maximally exposed off-site individual (MOI). A theoretical individual living at the DOE site or shipyard boundary receiving the maximum exposure. At the Savannah River Site, two separate MOI locations were evaluated depending upon whether the spent fuel facility is constructed on the Savannah River Site or is located at the existing Barnwell Nuclear Fuel Plant (hereafter referred to as the Barnwell Plant) which is adjacent to the Savannah River Site. At Hanford, two separate MOI locations were also evaluated depending upon whether a new facility is constructed in the 200 Area or modifications are made to the Fuels and Materials Examination Facility (FMEF) which is located in the 400 Area.

- Nearest public access individual (NPA). At larger DOE sites, highways used by the public may cross the federal reservation which includes the facility where naval spent nuclear fuel operations could be conducted. Consequently, these analyses included evaluation of the exposure to a theoretical motorist who might be stranded on such a highway at the time of an accident. Based on experience from emergency exercises, emergency response teams would be able to evacuate such an individual within 2 hours, so this was the exposure time used in the calculations. At naval shipyard locations, no public access highways exist, but military personnel, civilian employees, or their family members, including some who reside on the base, may be located outside the controlled industrial area boundary but inside the confines of the military base. Such personnel might be at their homes, in buildings, or on the roadways of the base at the time of an accident or at any time throughout the year for the evaluation of normal operations. The base residents are used as the NPA individuals at these shipyards for analyses of normal operations. In the event of a severe accident they would be evacuated within 2 hours under military control of the base, so this time was used in accident calculations. No NPA value was calculated for the Kesselring Site and the Nevada Test Site because there are no public roads which cross these sites, there are no residents, and there are no other public accesses.

- Maximally exposed individual at nearby communities is evaluated for accidents.

- General population within a 50-mile radius of the facility.

Exposure is calculated to result from direct radiation from the facility and exposure to radioactive contamination released to the air. Normal releases directly to the water pathway occur only at shipyards which are located directly on bodies of water, and contamination of the water at all sites results from fallout of airborne contamination. The releases to the air might result in exposure through several pathways described as follows:

- External direct exposure from immersion in the airborne radioactive material (air immersion)

- External direct exposure from radioactive material deposited on the ground (ground surface)

- Internal exposure from inhalation of radioactive aerosols and suspended particles (inhalation)

- Internal exposure from ingestion of terrestrial food and animal products (ingestion)

- Exposure from contaminated water (water release).

The radiation exposure is calculated by the computer programs discussed in Section F.1.3.6 in a manner recommended by the International Commission on Radiological Protection (ICRP 1977; ICRP 1979). Weighting factors are used for various body organs to calculate a "committed effective dose equivalent" (CEDE) from radiation inside the body due to inhalation or ingestion. Committed dose equivalents (CDEs) are calculated for organs such as the lungs, stomach, small intestine, upper large intestine, lower large intestine, bone surface red bone marrow, testes, ovaries, muscle, thyroid, bladder, kidneys, liver etc. The CEDE value is the summation of the CDEs to the specific organ weighted by the relative risk to that organ compared to an equivalent whole-body exposure.

The programs also calculate an effective dose equivalent (EDE) for the external exposure pathways (immersion in the radioactive material, exposure to ground contamination) and a 50-year CEDE for the internal exposure pathways . The sum of the EDE from external pathways and the CEDE internal pathways is called the "total effective dose equivalent" (TEDE) in this Environmental Impact Statement (EIS) and is also calculated by the programs. The TEDE reported in the results section is the sum of the TEDE's from air, water, and direct radiation exposures.

The exposure from ingestion of terrestrial food and animal products is calculated on a yearly basis. However, it is expected that continued consumption of contaminated food products by the public would be suspended after a Protective Action Guideline is reached. In 1991, the Environmental Protection Agency recommended protective action guidelines in the range of 1 to 5 rem whole-body exposure. To ensure a consistent analysis basis, no reduction of exposure due to a Protective Action Guideline was accounted for in the analysis. This would result in a conservative approach which may slightly overestimate health effects within an exposed population, but allows for consistent comparisons between alternatives.

Table F.1.3.2-1 identifies selected nearby communities for each site for which hypothetical exposures for a maximally exposed individual were calculated. In all cases, the MOI exposure was greater than maximum exposure at any nearby community. Calculations were performed for these localities to evaluate exposures for areas representative of the range of communities within 50 miles of the sites analyzed. The selection of these communities was not intended to indicate that other localities were not important. Other communities of interest in the vicinity of the sites in addition to those evaluated include a number of communities in Maine and New Hampshire near the Portsmouth Naval Shipyard, including Portsmouth, Durham, Eliot, Greenland, Kittery, New Castle, North Hampton, Ogunquit, Rye, and South Berwick.

Table F.1.3.2-1. Nearby communities for each site.

INEL	Howe, Atomic City, Arco, Blackfoot, Idaho Falls
Savannah River	Snelling, Barnwell, Jackson, Aiken, Allendale, Augusta, Sylvania, Bamberg, Wrens
Hanford	Othello, Richland, Prosser, Pasco, Yakima, Umatilla
Nevada Test Site	Beatty, Pahrump, Las Vegas
Oak Ridge	Oak Ridge, Harriman, Rockwood, Knoxville, Jefferson City
Puget Sound	Seattle, Tacoma, Olympia, Port Angeles
Pearl Harbor	Pearl City, Aiea, Pacific Palisades, Ewa Beach, Honolulu, Ewa, Wahiawa
Norfolk	Newport News, Hampton, Suffolk, Virginia Beach, Williamsburg
Portsmouth	Dover, Exeter, Hampton Beach, Sanford, Nashua, Lowell, Concord, Portland, Boston
Kesselring	Ballston Spa, Saratoga Springs, Amsterdam, Schenectady, Corinth

Table F.1.3.2-2 presents an example of the detailed exposure calculation results which were performed. The table shows the possible exposure pathways and individuals analyzed.

F.1.3.3 Evaluation of Health Effects. Health effects are calculated from the exposure results. The risk factors used for calculations of health effects are taken from Publication 60 of the International Commission on Radiological Protection (ICRP 1991). Table F.1.3.3-1 lists the appropriate factors used in the analysis of both the normal operations and the hypothetical accident scenarios.

Cancer fatalities were used to summarize and compare the results in this Environmental Impact Statement since this effect was viewed to be of the greatest interest to most people. As shown in Table F.1.3.3-1, the number of total health effects (deaths, non-fatal cancers, genetic effects, and other impacts on human health) may be easily obtained by multiplying the latent cancer fatalities by the factor of 1.46, which is the ratio of 7.3/5.0.

The numerical estimates of cancer deaths and other health detriments presented were obtained by the practice of linear extrapolation from the nominal risk estimate for lifetime total cancer mortality at 10 rad. Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of cancer deaths. Studies of human populations exposed at low doses are

Table F.1.3.2-2. Summary of exposure calculation results.

Location	Inhalation CEDE (rem)	Air Immersion EDE (rem)	Ground Surface EDE (rem)	Ingestion EDE (rem)	Airborne Release EDE (rem)	Water Release (rem)	Direct Radiation (rem)	Total EDE (rem)	Likelihood • of Fatal Cancer
Worker	5.4×10^{-1}	6.5×10^{-4}	7.9×10^{-1}	N/A	1.3	N/A	8.8×10^{-5}	1.3	5.3×10^{-4}
MCW	4.8×10^{-4}	8.6×10^{-7}	3.4×10^{-4}	N/A	8.2×10^{-4}	1.6×10^{-17}	3.8×10^{-8}	8.2×10^{-4}	4.1×10^{-7}
NPA	1.4×10^{-4}	3.2×10^{-7}	5.2×10^{-5}	N/A	1.9×10^{-4}	1.6×10^{-17}	3.4×10^{-9}	1.9×10^{-4}	9.5×10^{-8}
MOI	6.1×10^{-4}	1.2×10^{-6}	7.8×10^{-4}	3.1×10^{-4}	1.7×10^{-3}	3.0×10^{-5}	9.6×10^{-9}	1.7×10^{-3}	8.6×10^{-7}
Exposure to Maximally Exposed Individual at Nearby Communities (rem)									
Arco (30600m)	5.2×10^{-5}	1.3×10^{-7}	6.4×10^{-5}	3.1×10^{-5}	1.5×10^{-4}	3.0×10^{-5}	3.4×10^{-9}	1.8×10^{-4}	8.8×10^{-8}
Howe (16100m)	9.8×10^{-5}	1.8×10^{-7}	1.2×10^{-4}	5.6×10^{-5}	2.7×10^{-4}	3.0×10^{-5}	3.4×10^{-9}	3.0×10^{-4}	1.5×10^{-7}
Idaho Falls (72400m)	3.1×10^{-6}	5.2×10^{-9}	3.6×10^{-6}	2.0×10^{-6}	8.7×10^{-6}	3.0×10^{-5}	2.1×10^{-10}	3.9×10^{-5}	1.9×10^{-8}
Blackfoot (68100m)	4.8×10^{-6}	3.3×10^{-9}	5.2×10^{-6}	3.4×10^{-6}	1.3×10^{-5}	3.0×10^{-5}	2.1×10^{-10}	4.3×10^{-5}	2.2×10^{-8}
Atomic City (24200m)	2.9×10^{-5}	1.0×10^{-7}	3.6×10^{-5}	1.6×10^{-5}	8.1×10^{-5}	3.0×10^{-5}	3.4×10^{-9}	1.1×10^{-4}	5.6×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)									Fatal Cancers
Population of 115690	1.1×10^{-1}	6.1×10^{-5}	1.5×10^{-1}	4.5×10^{-2}	3.0×10^{-1}	3.8	5.3×10^{-6}	4.1	2.1×10^{-3}

213

Table F.1.3.3-1. Risk estimators for health effects from ionizing radiation.

Effect	Nuclide	Risk Factor (probability per rem)* Worker	General Population
Fatal cancer (all organs)	All	4.0×10^{-4}	5.0×10^{-4}
Weighted non-fatal cancer**	All	8.0×10^{-5}	1.0×10^{-4}
Weighted genetic effects**	All	8.0×10^{-5}	1.3×10^{-4}
Weighted total effects**	All	5.6×10^{-4}	7.3×10^{-4}

* For high individual exposures (≥20 rem), the above risk factors are multiplied by a factor of two. General population exposures were not modified because the large drop in exposure with increasing distances results in average exposure rates well below 20 rem.

** In determining a means of assessing health effects from radiation exposure, the ICRP has developed a weighting method for non-fatal cancers and genetic effects to obtain a total weighted effect, or "health detriment".

inadequate to demonstrate the actual level of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (CIRRPC 1992). In this appendix, the doses have been provided in all cases to allow independent evaluation using any relation between exposure and health effects.

F.1.3.4 Population. Population distributions specific to each site were used for the evaluations. The population distributions were obtained from 1990 United States Census data. The population information was obtained in 16 compass directions and 5 equal radial distances from the likely location of a naval spent nuclear fuel site to a 50-mile total distance.

F.1.3.5 Meteorology. For the navy shipyards, Savannah River, and Kesselring Sites, the meteorological data used in the analyses were obtained from the SCRAM bulletin board system. For the INEL, Hanford, Nevada Test Site, and Oak Ridge, site tower meteorological data were used. The SCRAM bulletin board is operated by the Support Center for Regulatory Air Models within the Environmental Protection Agency, Office of Air Quality Planning and Standards. The SCRAM surface meteorological data files are comprised of data acquired from the National Climatic Data Center. The SCRAM data for 4 or 5 years were used with programs from the bulletin board to develop meteorological data in the STability ARray (STAR) format which is a joint frequency distribution of 6 wind speed intervals, 16 wind directions, and 6 stability categories. The STAR data were reformatted into the format required by the GENII program, described below, for evaluation of normal operations.

The STAR data were also used to calculate the 50% and 95% meteorological conditions for the accident analyses. The 50% condition represents the average meteorological condition. This condition is defined as that for which more severe conditions with respect to accident consequences occur less than 50% of the time. The 95% condition represents the meteorological conditions which could produce the highest calculated exposures. This is defined as that condition which is not exceeded more than 5% of the time or is the worst combination of weather stability class and wind speed. Each of these conditions is evaluated for 16 wind directions.

For each location, the nearest available SCRAM data was used to represent the conditions at the site being evaluated. Table F.1.3.5-1 shows the pertinent data for the meteorological data application.

Table F.1.3.5-1. Meteorological data applicability.

Site	Data From	Data Years
Portsmouth	Portland ME Airport	1985-1989
Norfolk	Norfolk VA Airport	1985-1989
Puget Sound	SEATAC Airport	1985-1989
Pearl Harbor	Honolulu Airport	1985-1989
INEL	NRF Tower	1987-1991
Kesselring	Albany NY Airport	1985-1989
Savannah River	Augusta GA Airport	1984-1987
Hanford	200 Area Tower	1983-1990
Nevada Test Site	Desert Rock Tower	1990
Oak Ridge	Y-12 West Tower	1990

F.1.3.6 Computer Programs. Five computer programs were used to evaluate the radiation exposures to the specified individuals and general population.

F.1.3.6.1 GENII. The code used for the environmental and transport and exposure assessment calculations for normal operations was GENII (Napier et al. 1988). This code was developed at Pacific Northwest Laboratory by Battelle Memorial Institute to incorporate the internal dosimetry models recommended by the International Commission on Radiological Protection in Publication 26 (ICRP 1977) and Publication 30 (ICRP 1979) into environmental pathway analysis models in use at Pacific Northwest Laboratory.

Although GENII can be used to model both acute and chronic releases to the atmosphere, only the chronic option was used in the normal operations evaluation reflecting long-term average exposure to the released radioactive contaminants. For the chronic evaluations, the code also uses meteorological conditions averaged over each sector to reflect exposure to long-term average concentrations. The ingestion calculation used the modeling approach that exposed individuals within 50 miles of the site consumed 30% of milk products and 10% of all products grown locally where the people live.

F.1.3.6.2 RSAC-5. The computer code RSAC-5 was developed by Westinghouse Idaho Nuclear Co, Inc., for the DOE-ID Operations Office and is in the public domain (Wenzel 1993). The code calculates the consequences of the release of radionuclides to the atmosphere. It allows the amount of each fission product nuclide from a nuclear event to be input individually or to be calculated internally by the code. RSAC-5 calculates potential radiation exposures to maximally exposed individuals or population groups via inhalation, ingestion, exposure to radionuclides deposited on the ground surface, immersion in airborne radioactive material, and radiation from a cloud of radioactive material. RSAC-5 meteorological capabilities include Gaussian plume dispersion for Pascal-Gifford conditions. RSAC-5 release scenario modeling allows reduction of nuclides by chemical group or element and calculates decay and buildup during transport through operations, facilities, and the environment. It also models the effect of filters or other cleanup systems. Population exposures are the product of the calculated individual exposure and the number of people in the affected population.

F.1.3.6.3 ORIGEN. ORIGEN (Croff 1980) is a computer code system for calculating the buildup and decay of radioactive materials (fission products, actinides, and activation products). The code input was modeled to describe the naval nuclear fuel system and incorporates cross-section data that are distinct to naval fuels.

F.1.3.6.4 SPAN. SPAN (Wallace 1972) is the computer code which was used to calculate the direct radiation levels. Attenuation from air was included in the calculated radiation levels. To determine the unit person exposure per sector, SPAN was used to integrate the radiation level over the sector. The radiation levels calculated at various distances were used as the source to represent the proper distance falloff in the sector, and a total radiation level for each sector was calculated. This total integrated radiation level for each sector was then divided by the sector volume, resulting in an "average" radiation exposure for any point within the sector.

F.1.3.6.5 WATER RELEASE. WATER RELEASE is an unpublished computer code used to calculate exposures to humans arising from radionuclides which have been introduced into water in the vicinity of the proposed spent nuclear fuel storage and examination facilities. The following discussion provides a brief description of the key points associated with obtaining these estimates. All radionuclides which were considered to be introduced into the water at a site were postulated to be promptly distributed uniformly in the water in the immediate vicinity of the site during the time period in which the nuclides were introduced. There are two processes by which radionuclides might enter the water at each site: via liquid discharge or via airborne discharge. For liquid discharges, a fraction of the released radionuclides might enter the water accessed by humans each year by infiltrating the ground to the groundwater then traveling either to wells or surface water. For airborne discharges, some fraction of the released radionuclides might enter the water by deposition from the air. For both of these processes, the fraction of radionuclides that might enter the water used by humans has been postulated to enter the water immediately, except for NRF and the Nevada Test Site. For NRF and the Nevada Test Site, it has been postulated that 20 years pass before the nuclides might enter the water accessed by humans. This estimate is based upon the fact that water must percolate into the ground and reach groundwater resources. Further, contamination must travel with the water in the aquifer to a point where it can be used by humans, such as a well at Atomic City. An assessment of the infiltration rate of radionuclides beneath ICPP estimates that about 200 years are needed for them to pass into the aquifer (Smith 1994). Also, the water in the aquifer flows at a rate of 5 to 20 feet per day. Therefore, 20 years was used as the time for radionuclides to reach humans at INEL. Similarly, at the Nevada Test Site surface water is not present so water must reach aquifers which are more than 600 feet deep. Hence, 20 years was also used at this site.

Once the radionuclides have been introduced into the water at a site, they were calculated to be transported to locations where they might affect man either directly as via immersion (swimming) or indirectly as via ingestion of food. During this transport period, these radionuclides are subjected to various mechanisms which may reduce their concentration in the water such as radioactive decay, dilution in larger volumes of water, removal by sedimentation, etc. The pathways considered in this analysis by which radionuclides in the water at a site might reach man are immersion, exposure to surface deposits, boating and equipment exposure, and consumption of drinking water, fish, crustacea, molluscs, game animals, vegetables and fruits, root crops, milk and eggs, and domesticated animals. During the period when the radionuclides have left the water environment and are being transported through the pathways to man, they may be subjected to both concentration and removal mechanisms which will further modify their effect upon man. These mechanisms include concentration in the surface deposit, animal, and crop pathways; decay during periods between harvesting a crop and its ingestion by man; and removal of activity due to harvesting, handling, and cleaning of a foodstuff.

For each of the sites at which storage or examination of spent nuclear fuel is being considered, estimates were made for the exposures which the total population affected by releases from the site may receive and for the exposures which a maximally exposed individual may receive from these same releases. The exposures to the population affected at a given site were obtained by calculating the exposures received by an average individual in the vicinity of that site and multiplying that exposure by the number of people that are affected. The exposure to a maximally exposed individual used the maximum exposures and consumption rates which any individual at that site may experience regardless of the probabilities associated with just one individual actually following all the maximum pathways. The specific pathways which are applicable at a given site are dependent upon the site, since the exposure of an average or a maximum individual to each of the pathways is different for each of the sites. For example, exposures associated with the drinking water pathway are not considered for the shipyard sites since all radionuclides basically end up in salt water prior to their becoming available to man at these sites. On the other hand, the radionuclides introduced at the DOE and prototype sites can enter the drinking water pathway after a delay period. An initial delay occurs while the radionuclides seep through the ground soil before entering the aquifer. The delay continues while the radionuclides travel through the drinking water pathway and ultimately yield exposures to man. The total exposure to the population or to a maximally exposed individual at a given site is the resultant sum of the exposure commitments from the individual pathways applicable at that site.

F.1.3.7 Categorization of Accidents.

F.1.3.7.1 Abnormal Events. Abnormal Events are unplanned or improper events which result in little or no consequence. Abnormal events include industrial accidents and accidents during normal operations such as skin contamination with radioactive materials, spills of radioactive liquids, or exposure to direct radiation due to improper placement of shielding. The occurrence of these unplanned events has been anticipated and mitigative procedures are in place which promptly detect and eliminate the events and limit the effects of these events on individuals. As a result, there is little hazard to the general population from these events. Such events are considered to occur in the probability range of 1 to 10^{-3} per year. The probability referred to here is the total probability of occurrenc. .nd includes the probability the event occurs (e.g., plane crash) times other probabilities

required for the consequences. For accidents included in this range, results are presented for both the 50% meteorological condition (average meteorology) and the 95% meteorological condition.

F.1.3.7.2 Design Basis Accident Range. Accidents which have a probability of occurrence in the range of 10^{-3} to 10^{-6} per year are included in the range called the Design Basis Accident Range. The terminology "design basis accident," which normally refers to facilities to be constructed, also includes the "evaluation" basis accident which applies to existing facilities. For accidents included in this range, results are presented for both the 50% meteorological condition (average meteorology) and the 95% meteorological condition. Risk calculations for accidents in this range utilize the consequences associated with 95% meteorological conditions.

F.1.3.7.3 Beyond Design Basis Accidents. This range includes accidents which are less likely to occur than the design basis accidents but which may have very large or catastrophic consequences. Accidents included in this range typically have a total probability of occurrence in the range of 10^{-6} to 10^{-7} per year. Accidents which are less likely than 10^{-7} per year typically are not discussed since it is expected they do not contribute in any substantial way to the risk. For these beyond design basis accidents, consequences are presented for 50% and 95% meteorological conditions. Risk calculations for accidents in this range utilize the consequences associated with 95% meteorological conditions.

F.1.3.8 Evaluation of Impacted Area

The impacted area surrounding a facility following an accident was determined for each scenario evaluated. The impacted area was defined as that area in which the plume deposited radioactive material to such a degree that an individual standing on the boundary of the fallout area would receive approximately 0.01 mrem/hr of exposure. If this individual spends 24 hours a day at this location, that person would receive about 88 mrem per year from the ground surface shine. This is within the 100 mrem/year limit of 10CFR20.

To best characterize the affected areas for each casualty, a typical 50% meteorology was chosen (Pasquill-Gifford Class D, wind speed 10 mph) and applied to each accident scenario. The RSAC-5 results for ground surface dose were interpolated to determine the distance downwind where the centerline dose had dropped to approximately 88 mrem per year based on 24 hours per day exposure. For the wind class chosen, the plume remains within a singe 22.5-degree sector. The area affected by the plume is determined as the entire sector contaminated to the calculated downwind distance. Table F.1.3.8-1 lists each facility accident analyzed and the contaminated footprint associated with the accident.

F.1.3.8-1. Footprint estimates for facility accidents.

Accident Scenario	Footprint Length (miles)	Footprint Area* (acres)	Sites with Footprint Beyond Facility Boundary
Drained Water Pool	0.29	11	Norfolk, Oak Ridge, Portsmouth
Criticality	0.25	8	Norfolk, Oak Ridge, Portsmouth
Wet Storage Mechanical Damage	<0.06	<0.5	none
Wet Storage Airplane Crash	<0.06	<0.5	none
Dry Storage Mechanical Damage	<0.06	<0.5	none
Dry Storage Airplane Crash	0.91	106	Pearl Harbor, Norfolk, Oak Ridge, Portsmouth
Dry Cell Mechanical Damage	<0.06	<0.5	none
HEPA Filter Fire	<0.06	<0.5	none
Dry Cell Airplane Crash	1.27	207	Oak Ridge

*Based on contamination of a single sector.

Although the plume would be contained within a single sector, the direction of the wind is unknown. Therefore, each site was examined for impacts in all directions around the facility site out to a distance equal to the footprint length. Since the accidents do occur over a short duration of time, the acreage of the sector quoted is still an accurate indication of the total contaminated area. Identification of the potential impacts for each site is contained in Tables F.1.3.8-2 through -11.

Table F.1.3.8-2. Secondary impacts of facility accidents at Puget Sound Naval Shipyard.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Puget Sound Naval Shipyard	1. Dry Storage Plane Crash 2. Drained Water Pool 3. Criticality and all other radiological accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	Naval vessels at the shipyard could be temporarily contaminated during the accident. Cleanup operations would restore these ships to full readiness.	1. A total of approximately 106 acres might require cleanup. Contamination could extend about 0.6 miles beyond the closest site boundary. 2. Contamination might occur up to the nearest shipyard boundary but would be limited to approximately 10 acres total. 3.Contamination would be within the shipyard boundaries. Table F.1.3.8-1 lists the area that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found in Section 4.1.1 of this Appendix.	Access to some areas may be temporarily restricted until cleanup is completed. The total area restricted would be no greater than the areas identified under "Environmental Contamination".	No enduring impacts

Table F.1.3.8-3. Secondary impacts of facility accidents at Pearl Harbor Naval Shipyard.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Pearl Harbor Naval Shipyard	1. Dry Storage Plane Crash 2. All other radiological accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	Naval vessels at the shipyard could be temporarily contaminated during the accident. Cleanup operations would restore these ships to full readiness.	1. A total of approximately 106 acres might require cleanup. Contamination could extend about 0.4 miles beyond the closest site boundary. 2. Contamination would be within the shipyard boundaries. Table F.1.3.8-1 lists the areas that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found in Section 4.1.4 of this Appendix.	Access to some areas may be temporarily restricted until cleanup is completed. The total area restricted would be no greater than the areas identified under "Environmental Contamination".	No enduring impacts

222

Table F.1.3.8-4. Secondary impacts of facility accidents at Norfolk Naval Shipyard.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Norfolk Naval Shipyard	1. Dry Storage Plane Crash 2. Drained Water Pool and Criticality 3. All other radiological accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	Naval vessels at the shipyard could be temporarily contaminated during the accident. Cleanup operations would restore these ships to full readiness.	1. A total of approximately 106 acres might require cleanup. Contamination could extend about 0.8 miles beyond the closest site boundary. 2. This accident might contaminate about 10 acres which could extend beyond the nearest site boundary by about 0.1 miles 3. Contamination would be within the shipyard boundaries. Table F.1.3.8-1 lists the areas that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found in Section 4.1.2 of this Appendix.	Access to some areas may be temporarily restricted until cleanup is completed. The total area restricted would be no greater than the areas identified under "Environmental Contamination."	No enduring impacts

Table F.1.3.8-5. Secondary impacts of facility accidents at Portsmouth Naval Shipyard.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Portsmouth Naval Shipyard	1. Dry Storage Plane Crash 2. Drained Water Pool 3. Criticality and all other radiological accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	Naval vessels at the shipyard could be temporarily contaminated during the accident. Cleanup operations would restore these ships to full readiness.	1. A total of approximately 106 acres might require cleanup. Contamination could extend about 0.6 miles beyond the closest site boundary. 2. Contamination might occur up to the nearest shipyard boundary but would be limited to approximately 10 acres total. 3.Contamination would be within the shipyard boundaries. Table F.1.3.8-1 lists the areas that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found Section 4.1.3 of this Appendix.	Access to some areas may be temporarily restricted until clean-up is completed. The total area restricted would be no greater than the areas identified under "Environmental Consequences".	No enduring impacts

224

Table F.1.3.8-6. Secondary impacts of facility accidents at Oak Ridge Reservation.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Oak Ridge Reservation	1. Dry Cell Air Plane Crash 2. Dry Storage Plane Crash 3. Drained Water Pool and Criticality 4. All other radiological accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	No impacts	1. A total of approximately 207 acres might require cleanup. Contamination could extend about 1.1 miles beyond the closest site boundary. 2. This accident could contaminate about 106 acres and would extend beyond the nearest site boundary by about 0.7 miles. 3. About 10 acres might become contaminated extending about 0.1 miles offsite. 4. Contamination would remain within the site boundaries. Table F.1.3.8-1 lists the areas that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found in Section 4.5 of this Appendix.	Access to some areas may be temporarily restricted until cleanup is completed. The total area restricted would be no greater than the areas identified under "Environmental Consequences".	Some temporary restrictions on access may be required until cleanup is completed. No enduring impacts are expected.

Table F.1.3.8-7. Secondary impacts of facility accidents at Savannah River Site.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Savannah River Site	All Radiological Accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	No impacts	Contamination would remain within the site boundaries. Table F.1.3.8-1 lists the areas that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found in Section 4.3 of this Appendix.	Access to some areas may be temporarily restricted until cleanup is completed.	Some temporary restrictions on access may be required until cleanup is completed. No enduring impacts are expected.

226

Table F.1.3.8-8. Secondary impacts of facility accidents at Nevada Test Site.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Nevada Test Site	All Radiological Accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	No impacts	Contamination would remain within the site boundaries. Table F.1.3.8-1 lists the areas that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found Section 4.6 of this Appendix.	Access to some areas may be temporarily restricted until cleanup is completed.	Some temporary restrictions on access may be required until cleanup is completed. No enduring impacts are expected.

Table F.1.3.8-9. Secondary impacts of facility accidents at Idaho National Engineering Laboratory.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Idaho National Engineering Laboratory	All Radiological Accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	No impacts	Contamination would remain within the site boundaries. Table F.1.3.8-1 lists the areas that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found Section 4.2 of this Appendix.	Access to some areas may be temporarily restricted until cleanup is completed.	Some temporary restrictions on access may be required until cleanup is completed. No enduring impacts are expected.

228

Table F.1.3.8-10. Secondary impacts of facility accidents at Hanford Site.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Hanford Site	All Radiological Accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	No impacts	Contamination would remain within the site boundaries. Table F.1.3.8-1 lists the areas that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found Section 4.4 of this Appendix.	Access to some areas may be temporarily restricted until cleanup is completed.	Some temporary restrictions on access may be required until cleanup is completed. No enduring impacts are expected.

229

Table F.1.3.8-11. Secondary impacts of facility accidents at Kenneth A. Kesselring Site.

Site	Significant Accidents in Decreasing Severity	Biotic Resources	Water Resources	Economic Impacts	National Defense	Environmental Contamination	Endangered Species	Land Use	Treaty Rights
Kenneth A. Kesselring Site	1. Dry Storage Plane Crash 2. Drained Water Pool and all other radiological accidents	Plants and animals on the site and around the site will experience no long term impacts.	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected.	A small number of individuals may experience temporary job loss due to temporary restrictions on support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.	No impacts	1. Contamination is expected right up to the nearest site boundary but limited to approximately 106 acres total. 2.Contamination would remain within the shipyard boundaries. Table F.1.3.8-1 lists the areas that could be contaminated.	The facility accident would not result in the extermination of any species. Nor would it effect the long term potential for survival of any species. A listing of endangered species can be found Section 4.1.5 of this Appendix.	Access to some areas may be tempoarily restricted until cleanup is completed.	Some temporary restrictions on access may be required until cleanup is completed. No enduring impacts are expected.

230

F.1.3.9 Emergency Preparedness and Mitigative Measures.

F.1.3.9.1 Emergency Preparedness Emergency plans are in effect at shipyards and prototype sites to ensure that workers and the public would be properly protected in the event of an accident. In addition, emergency plans are in effect for accidents involving the transportation of radioactive materials. These response plans include the activation of emergency response teams provided by the site and a site emergency control center, as well as activation of a command and control network with Naval Reactors Headquarters and supporting laboratories. The long standing emergency planning program that exists within the Naval Nuclear Propulsion Program includes the ability to utilize the comprehensive and extensive emergency response resources of each naval site and provides for coordination with appropriate civil authorities. In addition to the Naval Nuclear Propulsion Program resources, extensive federal emergency response resources are available as needed to support State or local response.

Emergency response measures include provisions for immediate response to any emergency at the shipyard or prototype site, identification of the accident conditions, and communications with civil authorities providing radiological data and recommendations for any appropriate protective actions. In the event of an accident involving radioactive or toxic materials, workers in the vicinity of the accident would promptly evacuate the immediate area. This evacuation can typically be accomplished within minutes of the accident and would reduce the hazard to workers.

Regularly scheduled exercises are conducted periodically at each site in order to test each site's ability to respond to accidents. These exercises include realistic tests of people, equipment, and communications involved in all aspects of the plans, and the plans are regularly reviewed and modified to incorporate experience gained from the exercises. These exercises also periodically include steps to verify the adequacy of interactions with local hospitals and emergency personnel and state officials.

F.1.3.9.2 Mitigative Factors. For members of the general public residing at the site boundary or beyond, no credit is taken for any preventive or mitigative actions that would limit their exposure. These individuals are calculated as being exposed to the entire contaminated plume as it travels downwind from the accident site. Similarly no action is taken to prevent these people from continuing their normal day-to-day routine and ingestion of terrestrial food and animal products continue on a yearly basis. As discussed in Section F.1.3, action would be taken to prevent the

public from exceeding a Protective Action Guideline, if needed. No reduction of exposure due to these actions are accounted for in this analysis. The public is assumed to spend approximately 30% of the day within their homes or other buildings and the exposure to ground surface radiation is therefore reduced appropriately on a yearly basis.

Individuals that reside or work on site, or those that may be traversing the site in a vehicle would be evacuated from the affected area within 2 hours. This is based on the availability of security personnel at all locations to oversee the removal of residents, collocated workers, and travelers in a safe and efficient manner. Periodic training and evaluation of the security personnel is conducted to ensure that correct actions are taken during an actual casualty. Therefore, residents, collocated workers, and travelers would be exposed to the entire contaminated plume as it travels downwind for a period not to exceed 2 hours. Similarly, the radiation shine from the deposited radioactive materials would be limited to a 2-hour period. No ingestion of contamination is calculated for these individuals.

Facility workers all undergo training to take quick, decisive action during a casualty. These individuals quickly evacuate the area and move to previously defined "relocation" areas on the facility site. Workers could be exposed to a full 5 minutes of the radioactive plume as they move to the "relocation" centers. Once the immediate threat of the plume has moved off-site and downwind, the workers would be instructed to walk to vehicles waiting to evacuate them from the site. An additional 15 minutes would be required to evacuate the workers from the contaminated area and therefore the workers receive a total of 20 minutes of ground shine. No ingestion of contamination is calculated for these individuals.

The following summary provides the individual exposure times utilized in the accident analyses presented in Section F.1.4.2.

Estimated Time an Individual Might be Exposed

	Worker (100 m)	Collocated Worker (MCW) and Nearest Public Access (NPA)	Individual at Nearest Site Boundary (MOI)
To Plume	5 min.	100% of release time up to 120 min.	100% of release time
To Fallout on Ground Surface	20 min.	120 min.	0.7 yr
To Food	N/A	N/A	1 yr

F.1.3.10 Perspective on Calculations of Cancer Fatalities and Risk

The topics of human health effects caused by radiation and the risks associated with normal operations or postulated accidents associated with spent nuclear fuel management are discussed many times throughout this Environmental Impact Statement. It is important to understand these concepts and how they are used in order to understand the information presented in this document. It is also valuable to have some frame of reference or comparison for understanding how the risks compare to the risks of daily life.

The method used to calculate the risk of any impact is fundamental to all of the evaluations presented and follows standard accepted practices. The first step is to determine the probability that a specific event will occur. For example, the probability that a routine task, such as operating a crane, will be performed sometime during a year of normal operations at a facility would be 1. That means that the action would certainly occur. The probability that an accident might occur is less than 1.0. This is true because accidents occur only occasionally and some of the more severe accidents, such as a catastrophic earthquake, might occur at any location only once in hundreds, thousands, or millions of years.

Once the probability of an event has been determined, the next step is to predict what the consequences of the event being considered might be. One important measure of consequences chosen for this EIS is the number of human fatalities from cancer induced by radiation. This was chosen because this document deals with radioactive materials. The number of cancer fatalities that might be caused by any routine operation or any postulated accident can be calculated using a

standard technique based on the amount of radiation exposure that might occur from all conceivable pathways and the number of people who might be affected (refer to Section F.1.3.3).

A couple of examples should serve to illustrate the calculation of risk. In the first, the lifetime risk of dying in a motor vehicle accident can be computed from the likelihood of an individual being in an automobile accident and the consequences or number of fatalities per accident. There were 10,000,000 motor vehicle accidents during 1992 in the United States resulting in about 40,000 deaths (NSC 1993). Thus, the probability of a person being in an automobile accident is 10,000,000 accidents divided by approximately 250,000,000 persons in the United States, or 0.04 per year. The number of fatalities per accident, 0.004 (40,000 deaths divided by 10,000,000 accidents), is less than 1 since many accidents do not cause fatalities. Multiplying the probability of the accident (0.04 per year) by the consequences of the accident (0.004 deaths per accident) by the number of years the person is exposed to the risk (72 years is considered to be an average lifetime) gives the risk for any individual being killed in an automobile accident. From this calculation, the overall risk of someone dying in a motor vehicle accident is about 1 chance in 87 over their lifetime.

A second example illustrates the calculation of risk for another event which occurs daily. Fossil fuels, such as natural gas or coal, contain naturally occurring radioactive material that is released into the air during combustion. This radioactivity in the air finds its way into our bodies through our food and the air we breathe. This radioactivity has been estimated to produce about 0.5 millirem of radiation dose to the average American each year (NCRP 1987). The probability of this happening is essentially 1.0 since these fuels are burned every day all over the country. The number of fatal cancers from exposure to 0.5 millirem per year is calculated by taking 0.5 millirem per year times the 72 years considered to be an average lifetime times the 0.0005 fatal cancers estimated to be caused by each rem (0.5 millirem per year x 72 years x 0.0005 fatal cancers per rem = 0.000018 fatal cancers per individual lifetime). The risk is the probability (1.0) times the consequences (0.000018 cancer fatalities) which equals about 1 chance in 55,000 of death from this cause over a lifetime.

These risks and others from everyday life can be used to gain a perspective on the risks associated with the alternatives in this EIS. As illustrated, the risk of death from cancer from the radioactivity released daily from combustion of fossil fuels is about 1 chance in 55,000 for the average American. As a further comparison, the naturally occurring radioactive materials in agricultural fertilizer contribute about 1 to 2 millirem per year to an average American's exposure to radiation (NCRP 1987). A calculation similar to the one in the preceding paragraph shows that the use of fertilizer to produce food crops in the United States results in a risk of death from cancer between 1 chance in 12,500 and 1 chance in 25,000. Finally, the average American's risk of dying from cancer from all causes is 1 chance in 5 over his or her lifetime. These risks can be compared, for example, to the average individual risk of less than 1 chance in 1 billion for a resident in the vicinity of the INEL developing a fatal cancer due to normal operations at the Expended Core Facility (see the data in Section F.1.4.1).

A frame of reference for the risks from accidents associated with spent nuclear fuel management alternatives can be developed in the same way. For an average resident in the vicinity of the INEL, the individual risk of death from cancer caused by the water leaking from the Expended Core Facility after a large earthquake would be approximately 1 chance in 9 billion. This individual risk was determined by dividing the risk value to the population within 50 miles (1.7×10^{-7} fatalities per year per accident from Table F-3) by the total population of 115,690 and multiplying by an average life span of 72 years. This risk can be compared to the risks of death from other accidental causes to gain a perspective. For example, the risk of death in a motor vehicle accident was calculated earlier to be about 1 chance in 87. Similarly, the risk of death for the average American from fires is approximately 1 chance in 500, and for death from accidental poisoning the risk is about 1 chance in 1000 (Crouch 1982).

F.1.4 Analysis Results

F.1.4.1 Normal Operations. The purpose of this analysis is to determine the hypothetical health effects on workers and the public due to routine handling of naval spent nuclear fuel. Radioactive releases from facilities involved in routine handling of naval spent nuclear fuel are small and less than those of comparable DOE and commercial nuclear facilities. Records of routine releases due to operations at ECF were used as source terms for all locations to estimate what effects these types of releases have on workers and the public. Site-specific meteorological and population data were used at each of the locations analyzed. For normal operations at the Naval Reactors Facility (NRF and Oak Ridge), exposure to the nearest public access (NPA) individual is not estimated due to the short period of time that such an individual would spend on-site while driving on the public access road. At Hanford, the NPA is located at the Washington Public Power Supply System Plant, and at Savannah River at the U.S. Forestry Service Office. The NPA at shipyard locations is defined in Section F.1.3.2.

F.1.4.1.1 Water Pool Examination and Storage Source Terms. The evaluation of normal water pool operations was performed using two different source terms. In one analysis, a source term was utilized which included both the incremental release of radioactive materials due to the alternative spent nuclear fuel storage actions and the release from other ongoing Naval Reactors activities. Identical source terms were used for the evaluation of radiation exposure due to the release of radioactive materials during normal operations of wet storage and spent fuel examinations. The 1991 annual airborne release from the INEL-ECF was used to evaluate these operations. Since the INEL-ECF releases are extremely low, this upper limit approach is not unduly conservative for the wet storage option which is expected to have a lower release. Table F.1.4.1.1-1 shows the 1991 INEL-ECF release rate, the current release rate at Kesselring and NRF (including both INEL-ECF and prototypes), and the release rate representing Naval Reactors operations at naval shipyards. The release rate representing naval shipyards is based on upper bound data from Navy operations contained in Naval Nuclear Propulsion Program (NNPP) Report NT-94-1 (NNPP 1994). With no current Naval Reactors facilities at Savannah River, Hanford, Oak Ridge, or the Nevada Test Site, the current release for each of these sites is zero for this analysis.

Table F.1.4.1.1-1. Airborne releases from current Naval Reactors operations.

Location	Annual Releases (Ci/year)			
INEL-ECF	H-3	9.35×10^{-2}	Y-90	5.5×10^{-6}
	C-14	7.0×10^{-1}	I-131	4.82×10^{-6}
	Sr-90	5.5×10^{-6}	Kr-85	3.0×10^{-1}
NRF	H-3	9.35×10^{-2}	Sr-90	2.45×10^{-5}
	C-14	8.0×10^{-1}	Y-90	2.45×10^{-5}
	Ar-41	2.7×10^{-1}	I-131	6.3×10^{-6}
	Co-60	1.6×10^{-6}	Cs-137	6.3×10^{-6}
	Kr-85	3.0×10^{-1}		
Kesselring	H-3	1.0×10^{-1}	Kr-85	1.0×10^{-3}
	C-14	4.0×10^{-1}	I-131	5.0×10^{-4}
	Ar-41	1.4	Cs-137	5.0×10^{-4}
	Co-60	1.0×10^{-3}		
Savannah River, Hanford, Nevada Test Site, Oak Ridge	none			
Portsmouth, Norfolk Puget Sound, Pearl Harbor	H-3	1.0×10^{-3}	Kr-87	5.0×10^{-2}
	C-14	1.0×10^{-1}	Kr-88	2.0×10^{-2}
	Ar-41	4.1×10^{-1}	Xe131m	5.0×10^{-3}
	Co-60	1.0×10^{-3}	Xe133m	1.0×10^{-2}
	Kr-83m	2.0×10^{-2}	Xe-133	2.1×10^{-1}
	Kr-85m	2.4×10^{-2}	Xe-135	2.5×10^{-1}
	Kr-85	1.0×10^{-3}		

The evaluation of continuing Naval Reactors activities combined with the proposed alternatives for naval spent nuclear fuel is based on the combined airborne release source terms shown in Table F.1.4.1.1-2. This table presents a summation of the INEL-ECF source term and the current Naval Reactors operations source terms from Table F.1.4.1.1-1 for each location. Beginning in 1995, with the shutdown of the S5G prototype, the NRF releases will only result from the INEL-ECF, and this condition is shown in the table.

The other analysis utilized the same source term at all locations. The INEL-ECF source term of Table F.1.4.1.1-1 was used to compare the incremental health effects due to providing water pool storage or examination facilities at each location.

Both analyses also considered the impact on health effects of direct radiation levels from a water pool facility and the deposition of radionuclides onto the ground and into water supplies as discussed in Sections F.1.3.6.4 and F.1.3.6.5.

Table F.1.4.1.1-2. Airborne releases used in the analysis of water pool activities plus ongoing Naval Reactors operations.

Location	Annual Releases (Ci/year)			
NRF, Savannah River, Hanford, Nevada Test Site, Oak Ridge	H-3	9.35×10^{-2}	Y-90	5.5×10^{-6}
	C-14	7.0×10^{-1}	I-131	4.82×10^{-6}
	Sr-90	5.5×10^{-6}	Kr-85	3.0×10^{-1}
Kesselring	H-3	1.935×10^{-1}	Sr-90	5.5×10^{-6}
	C-14	1.1	Y-90	5.5×10^{-6}
	Ar-41	1.4	I-131	5.0×10^{-4}
	Kr-85	3.0×10^{-1}	Cs-137	5.0×10^{-4}
	Co-60	1.0×10^{-3}		
Portsmouth, Norfolk Puget Sound, Pearl Harbor	H-3	9.45×10^{-2}	Kr-88	2.0×10^{-2}
	C-14	8.0×10^{-1}	Sr-90	5.5×10^{-6}
	Ar-41	4.1×10^{-1}	Y-90	5.5×10^{-6}
	Co-60	1.0×10^{-3}	I-131	4.8×10^{-6}
	Kr-83m	2.0×10^{-2}	Xe131m	5.0×10^{-3}
	Kr-85m	2.4×10^{-2}	Xe133m	1.0×10^{-2}
	Kr-85	3.0×10^{-1}	Xe-133	2.1×10^{-1}
	Kr-87	5.0×10^{-2}	Xe-135	2.5×10^{-1}

F.1.4.1.2 Dry Storage Source Terms. Another operation analyzed was the storage of naval spent nuclear fuel in shipping containers or storage casks in a safe array at NRF, the naval shipyards, and Kesselring locations. It is postulated that shielding and physical boundaries are established in accordance with existing regulations to protect facility workers. There are expected to be no routine airborne or water releases from the dry storage activity. The source will consist of an array of filled storage containers. Supplementary shielding would be provided as needed to ensure that there would be no measurable increase in radiation levels at the perimeter of the industrial area and that radiation levels within the industrial area but outside the storage area would not require occupational radiation exposure monitoring for workers. Each location analyzed would have a different number of storage casks. As containers are received over time, shielding will be provided to limit radiation exposure rates as discussed above. Distance falloff for radiation levels was determined using SPAN computer calculations as discussed in Section F.1.3.6.4.

F.1.4.1.3 Dry Cell Facility Source Terms. The normal airborne release source terms utilized for the dry cell facility analyses are identical to the INEL-ECF releases in Table F.1.4.1-1. It is expected that these values bound the actual releases from the proposed facility. A source term different from the water pool analysis was utilized for the direct radiation calculations. This source term is based on the proposed facility design, expected fuel examination capacity, and shielding calculations. Like the airborne releases, source terms for water deposition were identical to those utilized in the water pool analysis.

F.1.4.1.4 Water Pool Storage. This section presents tabulated radiation exposure results for the wet storage option. The following summary provides an indication of the incremental change at each location due to the addition of an ECF-type facility.

Summary of Exposure Calculation Results
For Normal Operations - Water Pool Examination or Storage only
At All Sites

	INEL/NRF	Savannah River	Hanford	Puget Sound	Pearl Harbor
Worker EDE (rem)	7.1×10^{-5}	9.1×10^{-5}	8.9×10^{-5}	9.4×10^{-5}	1.1×10^{-4}
MOI EDE (rem)	2.5×10^{-7}	4.8×10^{-7} 3.8×10^{-6}*	2.4×10^{-7} 4.4×10^{-7}**	8.7×10^{-5}	2.0×10^{-5}
NPA EDE (rem)	N/A	2.1×10^{-8}	1.3×10^{-8}	6.2×10^{-4}	5.2×10^{-4}
Total EDE (person-rem)	1.7×10^{-3}	3.6×10^{-2}	8.0×10^{-3}	1.3×10^{-1}	1.4×10^{-1}
Number of Fatal Cancers	8.5×10^{-7}	1.8×10^{-5}	4.0×10^{-6}	6.5×10^{-5}	7.0×10^{-5}

* MOI (Barnwell Plant)
** MOI (FMEF)

	Norfolk	Portsmouth	Kesselring	Nevada Test Site	Oak Ridge
Worker EDE (rem)	6.9×10^{-5}	7.7×10^{-5}	8.5×10^{-5}	4.6×10^{-5}	1.2×10^{-4}
MOI EDE (rem)	1.1×10^{-4}	4.4×10^{-5}	6.8×10^{-6}	3.4×10^{-7}	1.0×10^{-4}
NPA EDE (rem)	6.8×10^{-5}	3.3×10^{-4}	N/A	N/A	N/A
Total EDE (person-rem)	2.8×10^{-1}	4.5×10^{-2}	8.2×10^{-2}	1.8×10^{-4}	1.0×10^{-1}
Number of Fatal Cancers	1.4×10^{-4}	2.3×10^{-5}	4.1×10^{-5}	9.0×10^{-8}	5.0×10^{-5}

Evaluations of environmental impacts at DOE sites are presented in Volume 1, Appendices A, B, C, and F. The radiological impacts at these sites are quite low in that fatal cancer projections to the population within 50 miles from normal operations are well below 1.0. Further, impacts at naval shipyards and prototype sites are addressed in Appendix D and also are well below 1.0. Hence, the addition of the above small values to those which already exist at a site result in total values which are also quite small.

The following summary provides the exposure calculation results for water pool storage or examination plus all ongoing Naval Reactors operations at each site.

Summary of Exposure Calculation Results
For Normal Operations - Water Pool Examination or Storage
plus all ongoing Naval Reactors operations
At all sites

	INEL/NRF	Savannah River	Hanford	Puget Sound	Pearl Harbor
Worker EDE (rem)	7.1×10^{-5}	9.1×10^{-5}	8.9×10^{-5}	1.2×10^{-4}	1.4×10^{-4}
MOI EDE (rem)	2.5×10^{-7}	4.8×10^{-7} 3.8×10^{-6}*	2.4×10^{-7} 4.4×10^{-7}**	1.0×10^{-4}	2.3×10^{-5}
NPA EDE (rem)	N/A	2.1×10^{-8}	1.3×10^{-8}	7.2×10^{-4}	5.8×10^{-4}
Total EDE (person-rem)	1.7×10^{-3}	3.6×10^{-2}	8.0×10^{-3}	1.5×10^{-1}	1.7×10^{-1}
Number of Fatal Cancers	8.5×10^{-7}	1.8×10^{-5}	4.0×10^{-6}	7.6×10^{-5}	8.5×10^{-5}

* MOI (Barnwell Plant)
** MOI (FMEF)

	Norfolk	Portsmouth	Kesselring	Nevada Test Site	Oak Ridge
Worker EDE (rem)	8.4×10^{-5}	9.7×10^{-5}	1.4×10^{-4}	4.6×10^{-5}	1.2×10^{-4}
MOI EDE (rem)	1.2×10^{-4}	5.0×10^{-5}	1.2×10^{-5}	3.4×10^{-7}	1.0×10^{-4}
NPA EDE (rem)	7.4×10^{-5}	3.5×10^{-4}	N/A	N/A	N/A
Total EDE (person-rem)	3.4×10^{-1}	5.5×10^{-2}	1.4×10^{-1}	1.8×10^{-4}	1.0×10^{-1}
Number of Fatal Cancers	1.7×10^{-4}	2.7×10^{-5}	7.2×10^{-5}	9.0×10^{-8}	5.0×10^{-5}

Tables F.1.4.1.4-1 through -10 present the detailed results of using the source terms of Table F.1.4.1-2 to determine the radiation exposures. These tables thus depict the result if an ECF-type examination operation is added to existing, current, continuing Naval Reactors operations at DOE sites and Navy shipyards.

Table F.1.4.1.4-1. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operations
At INEL

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.1×10^{-5}	2.8×10^{-8}
MCW	4.2×10^{-8}	1.7×10^{-11}
MOI	2.5×10^{-7}	1.3×10^{-10}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115,690	1.7×10^{-3}	8.5×10^{-7}

Table F.1.4.1.4-2. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operations
At Savannah River

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.1×10^{-5}	3.6×10^{-8}
MCW	1.4×10^{-6}	5.6×10^{-10}
MOI (New ECF)*	4.8×10^{-7}	2.4×10^{-10}
MOI (Barnwell Plant)**	3.8×10^{-6}	1.9×10^{-9}
NPA	2.1×10^{-8}	1.1×10^{-11}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579,541	3.6×10^{-2}	1.8×10^{-5}

* MOI (New ECF) applies if spent fuel facility is constructed on the Savannah River Site.
**MOI (Barnwell Plant) applies if spent fuel facility is constructed at Barnwell Nuclear Fuel Plant.

Table F.1.4.1.4-3. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operations
At Hanford

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.9×10^{-5}	3.6×10^{-8}
MCW	1.6×10^{-6}	6.4×10^{-10}
MOI (New ECF)*	2.4×10^{-7}	1.2×10^{-10}
MOI (FMEF)**	4.4×10^{-7}	2.2×10^{-10}
NPA	1.3×10^{-8}	6.5×10^{-12}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375,860	8.0×10^{-3}	4.0×10^{-6}

* MOI (New ECF) applies if spent fuel facility is constructed at the 200 area on the Hanford Site.
**MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Materials Examination Facility.

Table F.1.4.1.4-4. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operations
At Puget Sound

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.2×10^{-4}	4.8×10^{-8}
MOI	1.0×10^{-4}	5.1×10^{-8}
NPA	7.2×10^{-4}	3.6×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2,975,810	1.5×10^{-1}	7.6×10^{-5}

Table F.1.4.1.4-5. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operations
At Pearl Harbor

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.4×10^{-4}	5.6×10^{-8}
MOI	2.3×10^{-5}	1.1×10^{-8}
NPA	5.8×10^{-4}	2.9×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817,385	1.7×10^{-1}	8.5×10^{-5}

Table F.1.4.1.4-6. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operations
At Norfolk

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.4×10^{-5}	3.4×10^{-8}
MOI	1.2×10^{-4}	6.1×10^{-8}
NPA	7.4×10^{-5}	3.7×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1,539,002	3.4×10^{-1}	1.7×10^{-4}

Table F.1.4.1.4-7. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operations
At Portsmouth

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.7×10^{-5}	3.9×10^{-8}
MOI	5.0×10^{-5}	2.5×10^{-8}
NPA	3.5×10^{-4}	1.7×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2,432,627	5.5×10^{-2}	2.7×10^{-5}

Table F.1.4.1.4-8. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operations
At Kesselring

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.4×10^{-4}	5.6×10^{-8}
MOI	1.2×10^{-5}	5.8×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1,148,587	1.4×10^{-1}	7.2×10^{-5}

Table F.1.4.1.4-9. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operations
At Nevada Test Site

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.6×10^{-5}	1.8×10^{-8}
MCW	3.7×10^{-9}	1.5×10^{-12}
MOI	3.4×10^{-7}	1.7×10^{-10}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13,792	1.8×10^{-4}	9.0×10^{-8}

Table F.1.4.1.4-10. Summary of Exposure Calculation Results.
For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operations
At Oak Ridge

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.2×10^{-4}	4.8×10^{-8}
MCW	1.3×10^{-7}	5.1×10^{-11}
MOI	1.0×10^{-4}	5.1×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871,531	1.0×10^{-1}	5.0×10^{-5}

F.1.4.1.5 Dry Storage. This section presents tabulated radiation exposure results for the dry storage option at INEL, Navy shipyard sites, and the Kesselring Site. Dry storage at Hanford, Savannah River, the Nevada Test Site, and Oak Ridge is not included in this section as it is discussed in EIS Volume 1, Appendices A, C, and F, respectively. The following summary provides an indication of the incremental change at each location due to the addition of dry storage areas. The health effect due to dry storage of spent fuel is largest at the Navy shipyards and is extremely small at all DOE locations.

Summary of Exposure Calculation Results
For Normal Operations - Dry Storage only
At all sites

	INEL	Puget Sound	Pearl Harbor	Norfolk	Portsmouth	Kesselring
Worker EDE (rem)	1.1×10^{-2}	5.4×10^{-3}	2.1×10^{-3}	5.8×10^{-3}	2.7×10^{-3}	6.1×10^{-4}
MOI EDE (rem)	6.5×10^{-14}	8.9×10^{-5}	1.5×10^{-6}	2.9×10^{-3}	5.6×10^{-5}	5.2×10^{-11}
NPA EDE (rem)	N/A	7.4×10^{-3}	2.3×10^{-2}	2.9×10^{-3}	2.2×10^{-2}	N/A
Total EDE (person-rem)	1.7×10^{-12}	2.4×10^{-3}	1.9×10^{-5}	4.3×10^{-2}	4.6×10^{-4}	8.2×10^{-9}
Number of Fatal Cancers	8.6×10^{-16}	1.2×10^{-6}	9.3×10^{-9}	2.1×10^{-5}	2.3×10^{-7}	4.1×10^{-12}

Tables F.1.4.1.5-1 through -6 present the results if a dry storage area is added to existing, current, continuing Naval Reactors operations at all locations.

Table F.1.4.1.5-1. Summary of Exposure Calculation Results.
For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At INEL

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.1×10^{-2}	4.4×10^{-6}
MOI	1.1×10^{-10}	5.5×10^{-14}
NPA	6.5×10^{-14}	3.3×10^{-17}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115,690	1.7×10^{-12}	8.6×10^{-16}

Table F.1.4.1.5-2. Summary of Exposure Calculation Results.
For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Puget Sound

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.4×10^{-3}	2.2×10^{-6}
MOI	1.1×10^{-4}	5.3×10^{-8}
NPA	7.5×10^{-3}	3.8×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2,975,810	3.6×10^{-2}	1.8×10^{-5}

Table F.1.4.1.5-3. Summary of Exposure Calculation Results.
For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Pearl Harbor

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1×10^{-3}	8.5×10^{-7}
MOI	5.3×10^{-6}	2.7×10^{-9}
NPA	2.3×10^{-2}	1.2×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817,385	3.3×10^{-2}	1.7×10^{-5}

Table F.1.4.1.5-4. Summary of Exposure Calculation Results.
For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Norfolk

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.8×10^{-3}	2.3×10^{-6}
MOI	2.9×10^{-3}	1.5×10^{-6}
NPA	2.9×10^{-3}	1.5×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1,539,002	9.7×10^{-2}	4.9×10^{-5}

Table F.1.4.1.5-5. Summary of Exposure Calculation Results.
For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Portsmouth

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.7×10^{-3}	1.1×10^{-6}
MOI	6.3×10^{-5}	3.1×10^{-8}
NPA	2.2×10^{-2}	1.1×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2,432,627	9.2×10^{-3}	4.6×10^{-6}

Table F.1.4.1.5-6. Summary of Exposure Calculation Results.
For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Kesselring

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	6.6×10^{-4}	2.7×10^{-7}
MOI	5.1×10^{-6}	2.6×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1,148,587	5.7×10^{-2}	2.9×10^{-5}

F.1.4.1.6 Dry Cell Operations. This section presents tabulated radiation exposure results for the dry cell operations option. Since a facility like the proposed dry cell would only be constructed for the alternatives which include examination of all naval spent fuel, this analysis was only performed for the INEL, Savannah River, Hanford, the Nevada Test Site, and Oak Ridge locations. The following summary provides an indication of the incremental change at each location due to the addition of a dry cell facility. The calculated health effect to the general population is roughly proportional to the surrounding population with Oak Ridge being the worst and Nevada Test Site being the best.

Summary of Exposure Calculation Results
For Normal Operations - Dry Cell Operations
At all sites

	INEL/NRF	Savannah River	Hanford	Nevada Test Site	Oak Ridge
Worker EDE (rem)	6.3×10^{-5}	8.3×10^{-5}	8.1×10^{-5}	3.5×10^{-5}	1.1×10^{-4}
MOI EDE (rem)	2.5×10^{-7}	4.8×10^{-7} 3.8×10^{-6}*	2.4×10^{-7} 4.4×10^{-7}**	3.4×10^{-7}	8.9×10^{-5}
NPA EDE (rem)	N/A	2.1×10^{-8}	1.3×10^{-8}	N/A	N/A
Total EDE (person-rem)	1.7×10^{-3}	3.6×10^{-2}	8.0×10^{-3}	1.8×10^{-4}	1.0×10^{-1}
Number of Fatal Cancers	8.5×10^{-7}	1.8×10^{-5}	4.0×10^{-6}	9.0×10^{-8}	5.0×10^{-5}

* MOI (Barnwell Plant)
** MOI (FMEF)

Tables F.1.4.1.6-1 through -5 present the detailed analysis results.

Table F.1.4.1.6-1. Summary of Exposure Calculation Results.
For Normal Operations - Dry Cell Operations
At INEL

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	6.3×10^{-5}	2.5×10^{-8}
MCW	4.2×10^{-8}	1.7×10^{-11}
MOI	2.5×10^{-7}	1.3×10^{-10}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115,690	1.7×10^{-3}	8.5×10^{-7}

Table F.1.4.1.6-2. Summary of Exposure Calculation Results.
For Normal Operations - Dry Cell Operations
At Savannah River

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.3×10^{-5}	3.3×10^{-8}
MCW	1.3×10^{-6}	5.3×10^{-10}
MOI (New ECF)*	4.8×10^{-7}	2.4×10^{-10}
MOI (Barnwell Plant)**	3.8×10^{-6}	1.9×10^{-9}
NPA	2.1×10^{-8}	1.1×10^{-11}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579,541	3.6×10^{-2}	1.8×10^{-5}

* MOI (New ECF) applies if spent fuel facility is constructed on the Savannah River Site.
**MOI (Barnwell Plant) applies if spent fuel facility is constructed at Barnwell Nuclear Fuel Plant.

Table F.1.4.1.6-3. Summary of Exposure Calculation Results.
For Normal Operations - Dry Cell Operations
At Hanford

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.1×10^{-5}	3.2×10^{-8}
MCW	1.5×10^{-6}	6.1×10^{-10}
MOI (New ECF)*	2.4×10^{-7}	1.2×10^{-10}
MOI (FMEF)**	4.4×10^{-7}	2.2×10^{-10}
NPA	1.3×10^{-8}	6.5×10^{-12}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375,800	8.0×10^{-3}	4.0×10^{-6}

* MOI (New ECF) applies if spent fuel facility is constructed at the 200 area on the Hanford Site.
**MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Materials Examination Facility.

Table F.1.4.1.6-4. Summary of Exposure Calculation Results.
For Normal Operations - Dry Cell Operations
At Nevada Test Site

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.5×10^{-5}	1.5×10^{-8}
MCW	3.7×10^{-9}	1.5×10^{-12}
MOI	3.4×10^{-7}	1.7×10^{-10}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13,792	1.8×10^{-4}	9.0×10^{-8}

Table F.1.4.1.6-5. Summary of Exposure Calculation Results.
For Normal Operations - Dry Cell Operations
At Oak Ridge

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.1×10^{-4}	4.4×10^{-8}
MCW	1.1×10^{-7}	4.6×10^{-11}
MOI	8.9×10^{-5}	4.5×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871,531	1.0×10^{-1}	5.0×10^{-5}

F.1.4.2 Accident Evaluation. The analysis of airborne releases from hypothetical accidents is evaluated with RSAC-5. Unless stated otherwise, the following conditions were used when performing calculations with RSAC-5. In most cases, these conditions are taken directly as defaults from the code.

Meteorological Data

- Wind speed, direction, and Pasquill stability are taken from 50% and 95% meteorology. See Section F.1.3.5 for a discussion of meteorological conditions.

- The release is calculated as occurring at ground level (0 m).

- Mixing layer height is 400 meters (1320 feet). Airborne materials freely diffuse in the atmosphere near ground level in what is known as the mixing depth. A stable layer exists above the mixing depth which restricts vertical diffusion.

- Wet deposition is zero (no rain occurs to accelerate deposition and reduce the area affected).

- Dry deposition of the cloud is modeled. During movement of the radioactive plume, a fraction of the plume is deposited on the ground due to gravitational forces and becomes available for exposure by ground surface radiation and ingestion.

- The quantity of deposited radioactive material is proportional to the material size and speed. The following dry deposition velocities (m/s) were used:
  solids = 0.001 halogens = 0.01 noble gases = 0.0
  cesium = 0.001 ruthenium = 0.001.

- If radioactive releases occur through a stack, then additional plume dispersion can be accounted for by calculating a jet plume rise. In this analysis, jet plume rise is ignored.

- When released gases have a heat content, the plume can disperse more quickly. In this calculation, buoyant plume effects are ignored.

Inhalation Data

- Breathing rate is 3.33 x 10^{-4} cubic meters per second (cu m/s) for worker, MCW, and NPA; 2.66 x 10^{-4} cu m/s for people at site boundary and beyond.
- Particle size is 1.0 micron.
- The internal exposure period is 50 years for individual organs and tissues which have radionuclides committed.
- Exposure to the entire plume for the general public. The worker, MCW, and NPA are exposed as discussed in Section F.1.3.9.
- Inhalation exposure factors based on ICRP 30.

Ground Surface Exposure

- Exposed to contaminated soil for 1 year for the general public. See Section F.1.3.9 for additional details.
- Building shielding factor is 0.7 which exposes the individual to contaminated soil for 16 hours a day.

Ingestion Data

- Ingestion numbers will be reduced by a factor of 10 to account for only 10% of the food consumed being grown locally (such as in a person's garden).
- The following changes from RSAC-5 defaults were used:

  Annual Dietary Consumption Rates:

  177 Kg/yr Stored Vegetables (produce)
  18.3 Kg/yr Fresh Vegetables (leafy)
  94 Kg/yr Meat
  112 L/yr Milk.

F.1.4.2.1 Water Pool Storage. In the analysis of a spent fuel storage pool, a number of possible disturbances and minor accidents have been postulated. A prerequisite for a large release of radioactive material to the environment under more severe accident conditions is the damage of the cladding of a fairly large amount of stored fuel, with an accompanying release of gaseous and airborne particles of radioactive material from the fuel. Several conceivable mechanisms which might lead to this situation are the possibility that the fuel overheats so that the fuel cladding loses its integrity or there is a massive mechanical impact on the stored fuel.

The only way for the fuel to overheat would be to lose enough pool water such that cooling of the stored fuel ceases and the fuel temperature increases to fission product release temperatures due to decay heat. The pool water could be lost by leakage at a rate in excess of the makeup system capability. Unless a catastrophic event like an earthquake causes severe damage to the structure of the water pool, loss of water from the pool structure would be a slow phenomenon with only gradually increasing severity for which corrective measures can be taken in due time. Additionally, a thermal analysis was conducted to demonstrate that fuel overheating is not possible in the event of a drained water pool.

The circumstances in which an event could lead to severe mechanical loading of the fuel have been identified as:

- accidents during handling of heavy items, such as a lifting device failure
- external events (earthquake, tornado, flood, aircraft crash, etc.) which could cause structural failure.

Prevention of inadvertent, uncontrolled nuclear chain reactions is generally assured by the design of the racks for the fuel, primarily by diminishing the chances for a chain reaction by spacing the fuel element bundles far enough apart to eliminate the possibility. Special attention is given to the risk of accidental criticality which might be experienced in fuel transport and handling operations. Uncontrolled nuclear reaction is prevented during fuel handling by applying the principle of transferring one fuel element, module, or container at a time. In addition, fuel handling rules are developed to ensure that criticality cannot occur. The double accident criterion is applied to ensure that criticality would not occur following two severe, concurrent, unrelated accidents. Thus, three fuel handling accidents are required to reach an uncontrolled nuclear chain reaction.

F.1.4.2.1.1 Drained Water Pool.

F.1.4.2.1.1.1 Description of Conditions. In this hypothetical accident scenario, a catastrophic event, like an earthquake, causes severe damage to the structure of the water pool, resulting in a complete loss of pool water. A thermal analysis of spent fuel in a water pool was conducted to demonstrate that clad failure or fuel melting is not possible in the event of an accidentally drained water pool. Air circulation through the fuel racks and fuel units was shown to be sufficient to prevent clad failure in the unlikely event of complete loss of pool water. However, the loss of water could result in increased direct radiation and a release of corrosion products.

F.1.4.2.1.1.2 Source Term. Conditions used in developing the source term are as follows:

- 300 naval fuel units would be in the water pool.
- The thermal analysis demonstrates that no fission product release would occur during the accident.
- The amount of corrosion products on the fuel units is based on best estimate values.
- The release to the environment would occur at a constant rate over a 15-minute period.
- One percent of the original corrosion products from the fuel units might be released to the atmosphere due to thermal air currents. Additionally, 10% of the corrosion products could be released to the environment with the pool water.
- The following amounts of corrosion product nuclides might be released to the atmosphere. As noted above, the release to the water environment is 10 times these values. This listing includes nuclides that result in at least 99% of the exposure.
- No filtration by High Efficiency Particulate Air (HEPA) filters is assumed.

Nuclide	Curies
Co-60	3.6
Fe-55	6.6
Co-58	1.3
Mn-54	2.2×10^{-1}
Fe-59	1.9×10^{-2}

F.1.4.2.1.1.3 Results. The following table summarizes the public health risk to the general population that might result from the hypothetical drained water pool accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. The results are presented for the design basis accident with 50% and 95% meteorology. For INEL, the evaluation basis earthquake results in a 0.24 g peak ground acceleration at the ECF (Rizzo 1994). This is based on the event being initiated at the Howe earthquake epicenter and involving a surface rupture length of 34 kilometers. Using the medium response spectra, which is appropriate for a risk oriented analysis, the analyses of the structures at the INEL-ECF indicate that damage sufficient to cause the pool to drain would not occur if the pool is filled, but that, if several sections of the water pool were empty, a crack could develop in the area between the wall and floor of some of the older sections of the water pool. However, the INEL-ECF water pools are nearly always filled. Sections of the pool are only drained if maintenance work is necessary within the pools. Taking into account the probability of the initiating seismic event (1×10^{-4} per year to 4×10^{-4} per year) and the probability the earthquake will occur with a section of the pool drained, the total probability of occurrence of an event leading to draining of the pool is estimated to be in the range of 10^{-5} to 10^{-6} per year. A value of 10^{-5} was used to develop the risk results in the table.

A beyond design basis seismic event was also considered. For INEL, this beyond design basis earthquake is based on a scenario that results in a peak ground acceleration at the INEL-ECF of 0.40 g (Rizzo 1994). Analysis of this event has shown that some cracks could develop. The probability of this beyond design basis event is estimated to be in the range of 10^{-6} to 10^{-7} per year based on the probability of the initiating seismic event (2×10^{-5} to 6×10^{-5}), and the probability of failure of the mitigative actions that would be taken to prevent the pool from draining. A value of 10^{-6} was selected to calculate risk for this beyond design basis event. Any cracks developed as a result of either a design basis or a beyond design basis seismic event are expected to be small and mitigative actions could be taken to stop the pool from draining. Analysis has shown that air cooling is sufficient to maintain fuel integrity if the pool was drained. No overheating of fuel would occur; hence, no fission products would be released even if the pool were completely drained. The consequences calculated stem from the release of radioactive corrosion products within the pool water and would be the same for the design basis and beyond design basis seismic events. Since the consequences are the same, the following table uses the accident probability for the design basis seismic event since that results in the larger risk.

For locations other than INEL, water pools might need to be constructed. For these locations, it was expected that the design approaches would be similar to or better than were used in the construction of the INEL-ECF. Therefore, a probability value of 10^{-5} per year was also used at these locations for the total probability that a design basis seismic event would lead to draining of a water pool. Consequences were based on site specific population data and meteorology.

Drained Water Pool Summary			
Site	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
INEL	1.7×10^{-2}	1.7×10^{-2}	1.7×10^{-7}
Savannah River	1.6×10^{-2}	1.1×10^{-1}	1.1×10^{-6}
Hanford	6.3×10^{-3}	4.7×10^{-2}	4.7×10^{-7}
Puget Sound	1.4	5.1×10^{-1}	5.1×10^{-6}
Pearl Harbor	7.9×10^{-1}	1.1	1.1×10^{-5}
Norfolk	3.0	6.0×10^{-1}	6.0×10^{-6}
Portsmouth	1.6	3.4×10^{-1}	3.4×10^{-6}
Kesselring	2.9×10^{-1}	2.5×10^{-1}	2.5×10^{-6}
Nevada Test Site	3.3×10^{-2}	1.9×10^{-3}	1.9×10^{-8}
Oak Ridge	5.2	1.8×10^{-1}	1.8×10^{-6}

The risk for this hypothetical accident is generally more severe at Navy shipyards than at the DOE sites. At all sites, this accident results in the highest risk of the wet storage accidents evaluated.

For the hypothetical drained water pool scenario, the radioactive plume might result in contamination of the ground to a downwind distance of 0.29 mile. This would yield a total area impacted by the accident of approximately 11 acres. The calculated downwind distance would be contained within the boundaries of all sites under evaluation with the exception of Oak Ridge and Norfolk.

Table F.1.4.2.1.1-1. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At INEL

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.5×10^{-1}	3.0×10^{-4}
MCW	6.9×10^{-4}	2.7×10^{-7}
NPA	3.9×10^{-4}	2.0×10^{-7}
MOI	2.8×10^{-3}	1.4×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	6.7	3.3×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	7.6×10^{-3}	3.0×10^{-6}
NPA	2.3×10^{-3}	1.2×10^{-6}
MOI	1.7×10^{-2}	8.5×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	3.5×10^{1}	1.7×10^{-2}

Table F.1.4.2.1.1-2. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Savannah River

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.4×10^{-1}	1.3×10^{-4}
MCW	2.0×10^{-2}	7.9×10^{-6}
NPA	2.5×10^{-4}	1.3×10^{-7}
MOI (New ECF)	3.5×10^{-3}	1.8×10^{-6}
MOI (Barnwell)	1.3×10^{-2}	6.3×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	2.4×10^{1}	1.2×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	2.5×10^{-1}	1.0×10^{-4}
NPA	4.3×10^{-3}	2.1×10^{-6}
MOI (New ECF)	1.6×10^{-2}	8.0×10^{-6}
MOI (Barnwell)	1.4×10^{-1}	7.2×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	2.2×10^{2}	1.1×10^{-1}

Table F.1.4.2.1.1-3. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Hanford

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.4×10^{-1}	1.3×10^{-4}
MCW	2.6×10^{-2}	1.0×10^{-5}
NPA	3.0×10^{-4}	1.5×10^{-7}
MOI (New ECF)	8.3×10^{-4}	4.2×10^{-7}
MOI (FMEF)	1.7×10^{-3}	8.6×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	4.8	2.4×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	1.6×10^{-1}	6.6×10^{-5}
NPA	4.8×10^{-3}	2.4×10^{-6}
MOI (New ECF)	6.3×10^{-3}	3.2×10^{-6}
MOI (FMEF)	2.2×10^{-2}	1.1×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	9.4×10^{1}	4.7×10^{-2}

Table F.1.4.2.1.1-4. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Puget Sound

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.8×10^{-1}	7.3×10^{-5}
MCW	N/A	N/A
NPA	2.2×10^{-1}	1.1×10^{-4}
MOI	1.2×10^{-1}	6.0×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	1.7×10^{2}	8.2×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	N/A	N/A
NPA	2.6	1.3×10^{-3}
MOI	1.4	7.2×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	1.0×10^{3}	5.1×10^{-1}

Table F.1.4.2.1.1-5. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Pearl Harbor

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.5×10^{-1}	3.0×10^{-4}
MCW	N/A	N/A
NPA	1.9×10^{-1}	9.7×10^{-5}
MOI	2.0×10^{-1}	9.8×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	8.0×10^{2}	4.0×10^{-1}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	N/A	N/A
NPA	6.3	3.1×10^{-3}
MOI	7.9×10^{-1}	3.9×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	2.2×10^{3}	1.1

Table F.1.4.2.1.1-6. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Norfolk

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.8×10^{-1}	7.4×10^{-5}
MCW	N/A	N/A
NPA	4.6×10^{-2}	2.3×10^{-5}
MOI	2.8×10^{-1}	1.4×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	1.5×10^{2}	7.7×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	N/A	N/A
NPA	5.3×10^{-1}	2.7×10^{-4}
MOI	3.0	1.5×10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	1.2×10^{3}	6.0×10^{-1}

Table F.1.4.2.1.1-7. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Portsmouth

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.8×10^{-1}	7.3×10^{-5}
MCW	N/A	N/A
NPA	4.4×10^{-2}	2.2×10^{-5}
MOI	1.3×10^{-1}	6.4×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	6.5×10^{1}	3.2×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	N/A	N/A
NPA	9.8×10^{-1}	4.9×10^{-4}
MOI	1.6	7.9×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	6.7×10^{2}	3.4×10^{-1}

Table F.1.4.2.1.1-8. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Kesselring

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.8×10^{-1}	7.4×10^{-5}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	2.0×10^{-2}	1.0×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	7.1×10^{1}	3.6×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	2.9×10^{-1}	1.5×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	5.0×10^{2}	2.5×10^{-1}

Table F.1.4.2.1.1-9. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Nevada Test Site

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.2×10^{-1}	4.8×10^{-5}
MCW	9.3×10^{-5}	3.7×10^{-8}
NPA	N/A	N/A
MOI	1.5×10^{-3}	7.5×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	3.2×10^{-1}	1.6×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	5.4×10^{-3}	2.2×10^{-6}
NPA	N/A	N/A
MOI	3.3×10^{-2}	1.7×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	3.7	1.9×10^{-3}

Table F.1.4.2.1.1-10. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Oak Ridge

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.5×10^{-1}	3.0×10^{-4}
MCW	2.0×10^{-2}	7.9×10^{-6}
NPA	2.6×10^{-1}	1.3×10^{-4}
MOI	8.2×10^{-1}	4.1×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	7.1×10^{1}	3.6×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1	8.3×10^{-4}
MCW	1.2×10^{-1}	4.8×10^{-5}
NPA	1.6	8.2×10^{-4}
MOI	5.2	2.6×10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	3.5×10^{2}	1.8×10^{-1}

F.1.4.2.1.2 Accidental Criticality.

F.1.4.2.1.2.1 Description of Conditions. In this hypothetical accident scenario, an accidental uncontrolled chain reaction producing 1×10^{19} fissions is postulated. The criticality occurs in the water pool which is not emptied by the event and does not subsequently empty. Release of fission products includes those specified in Regulatory Guide 3.34 (NUREG 1979b) from the criticality, plus fission products remaining in the fuel as a result of the original use. Removal of fission products by the pool water is included.

F.1.4.2.1.2.2 Source Term. Conditions used in developing the source term are as follows:

- The fraction of the fission products released to the building is 100% of the noble gases, 25% of the halogens, 0.1% of the ruthenium (Elder et al. 1986), and 0.05% of the cesium and remaining solids.
- The original inventory of fission products from two naval fuel units are available for release in addition to those created by the criticality event.
- A High Efficiency Particulate Air (HEPA) filter removes 99.9% of the solid fission products from the plume.
- The release to the environment occurs at a constant rate over a 15-minute period. This is conservative as compared to the 8-hour release allowed in Regulatory Guide 3.34.

- The following amounts of radionuclides are released to the environment. This listing includes nuclides that result in at least 99% of the possible exposure.

Nuclide	Curies	Nuclide	Curies
Te-133	3.4×10^{3}	I-132	1.7×10^{0}
I-134	3.5×10^{2}	Sr-90	1.94×10^{-2}
I-135	1.2×10^{2}	Y-91m	4.3×10^{-8}
Cs-138	1.6×10^{-4}	Rb-88	1.7×10^{-5}
Rb-89	6.05×10^{-4}	Y-91	1.1×10^{-2}
Pu-238	3.7×10^{-4}	Cs-139	7.3×10^{-3}
Br-84	2.3×10^{2}	Ba-142	4.8×10^{-3}
I-133	2.4×10^{0}	Y-93	1.3×10^{-6}
Sr-91	5.4×10^{-6}	Ba-137m	1.9×10^{-2}
Sr-92	2.4×10^{-4}	Ru-106	7.6×10^{-3}
Ba-139	6.9×10^{-6}	Zr-95	1.4×10^{-2}
Ba-141	8.8×10^{-4}	Sr-89	7.01×10^{-3}
I-129	5.1×10^{-3}	Eu-154	1.3×10^{-3}
I-131	3.2×10^{-1}		
H-3	1.42×10^{2}		
Cs-134	1.5×10^{-2}		
Ba-140	2.5×10^{-5}		
I-136	1.1×10^{4}		
Cs-137	2.0×10^{-2}		
Ce-144	4.5×10^{-2}		
Nb-95	2.7×10^{-2}		
Rb-90	2.2×10^{-2}		

F.1.4.2.1.2.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical criticality accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. An accidental criticality during spent nuclear fuel handling operations is extremely unlikely. There are no known events of this type which have occurred during handling of fuel modules either in or out of water. Due to the need for a neutron moderator, extremely large quantities of naval fuels would be required to achieve criticality in a dry state. Fuel handling procedures in water in conjunction with required physical barriers ensure that a double accident criterion is met. This criterion specifies that the fuel will not attain a critical condition even if any two unlikely and unrelated accidents occur at the same time. The DOE criticality control requirement is a double contingency criterion which specifies that a second unlikely and unrelated accident would be required for a critical condition to result. To satisfy the NNPP double accident criterion, naval fuel handling operations are conducted in the following manner:

- No more than one module is to be handled in one area at a time.
- If two modules are capable of achieving a critical condition, separation must be maintained by a positive barrier between them which is locked in place.
- If three modules are required to achieve criticality, a physical barrier which does not need to be locked is required to be placed between them.
- If four or more modules are needed to achieve criticality, no barriers are required, but modules are to remain separated.

Based on the above requirements, at least three distinct errors are needed to achieve accidental criticality. For example, bringing two or more modules in close proximity is always prohibited. Failure to maintain separation constitutes an error. Secondly, failure to recognize and use physical barriers when required also constitutes an error. A human error rate of 10^{-3} per operation (Swain and Guttman 1983) is taken as the probability of error for trained personnel. Further, because all fuel handling operations must be checked by an independent verifier, an additional factor of 10^{-1} may be taken for a probability of 10^{-4} for each independent error. For naval fuel handling, an error in which two modules are brought together is a violation of a fundamental requirement. Compliance with this requirement alone ensures that a subcritical state is maintained. Therefore, the bringing of two or more modules together error is considered separate and independent of all other errors. Because a second error must occur to cause accidental criticality, an additional reduction in the probability is warranted. For example, failure to recognize the need to install a barrier when required is such an error. Because this mistake is independent of the first error and has been checked, a second value of 10^{-4} is appropriate for a total value of 10^{-8} per year. This probability is taken as the likelihood of a criticality for movement of a single module. Based on an estimated 1,000 fuel handling operations a year, a value of 10^{-5} per year has been used in the risk assessment of accidental criticality.

Accidental Criticality Summary			
Site	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
INEL	9.2×10^{-3}	6.4×10^{-3}	6.4×10^{-8}
Savannah River	9.4×10^{-3}	4.5×10^{-2}	4.5×10^{-7}
Hanford	2.8×10^{-3}	1.6×10^{-2}	1.6×10^{-7}
Puget Sound	1.3	2.8×10^{-1}	2.8×10^{-6}
Pearl Harbor	6.7×10^{-1}	6.0×10^{-1}	6.0×10^{-6}
Norfolk	2.7	3.5×10^{-1}	3.5×10^{-6}
Portsmouth	1.4	1.5×10^{-1}	1.5×10^{-6}
Kesselring	2.3×10^{-1}	1.1×10^{-1}	1.1×10^{-6}
Nevada Test Site	2.0×10^{-2}	7.0×10^{-4}	7.0×10^{-9}
Oak Ridge	4.7	8.8×10^{-2}	8.8×10^{-7}

The risk for this hypothetical accident is more severe at Navy shipyards than at the DOE sites. At all sites, this accident results in the second highest risk of the wet storage accidents evaluated.

For the hypothetical criticality accident scenario, the radioactive plume might cause contamination of the ground to a downwind distance of 0.25 mile. This would yield a total area impacted by the accident of approximately 8 acres. The calculated downwind distance would be contained within the boundaries of all sites under evaluation with the exception of Oak Ridge and Norfolk.

Table F.1.4.2.1.2-1. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At INEL

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.0	1.2×10^{-3}
MCW	1.3×10^{-3}	5.1×10^{-7}
NPA	5.9×10^{-4}	2.9×10^{-7}
MOI	2.0×10^{-3}	1.0×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	5.5	2.8×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2×10^{-3}
MCW	1.3×10^{-2}	5.0×10^{-6}
NPA	2.8×10^{-3}	1.4×10^{-6}
MOI	9.2×10^{-3}	4.6×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	1.3×10^{1}	6.4×10^{-3}

Table F.1.4.2.1.2-2. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Savannah River

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.3	5.3×10^{-4}
MCW	6.8×10^{-2}	2.7×10^{-5}
NPA	7.4×10^{-4}	3.7×10^{-7}
MOI (New (ECF)	3.3×10^{-3}	1.6×10^{-6}
MOI (Barnwell)	1.2×10^{-2}	5.9×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	2.2×10^{1}	1.1×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2×10^{-3}
MCW	7.9×10^{-1}	3.1×10^{-4}
NPA	6.4×10^{-3}	3.2×10^{-6}
MOI (New ECF)	9.4×10^{-3}	4.7×10^{-6}
MOI (Barnwell)	1.1×10^{-1}	5.3×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	8.9×10^{1}	4.5×10^{-2}

Table F.1.4.2.1.2-3. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Hanford

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.3	5.3×10^{-4}
MCW	8.9×10^{-2}	3.5×10^{-5}
NPA	6.6×10^{-4}	3.3×10^{-7}
MOI (New (ECF)	4.7×10^{-4}	2.4×10^{-7}
MOI (FMEF)	1.3×10^{-3}	6.7×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	2.2	1.1×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2×10^{-3}
MCW	4.9×10^{-1}	2.0×10^{-4}
NPA	6.9×10^{-3}	3.5×10^{-6}
MOI (New ECF)	2.8×10^{-3}	1.4×10^{-6}
MOI (FMEF)	1.2×10^{-2}	6.1×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	3.1×10^{1}	1.6×10^{-2}

Table F.1.4.2.1.2-4. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Puget Sound

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.2×10^{-1}	2.9×10^{-4}
MCW	N/A	N/A
NPA	7.7×10^{-1}	3.8×10^{-4}
MOI	1.1×10^{-1}	5.6×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	2.3×10^{2}	1.1×10^{-1}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2×10^{-3}
MCW	N/A	N/A
NPA	8.8	4.4×10^{-3}
MOI	1.3	6.3×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	5.6×10^{2}	2.8×10^{-1}

Table F.1.4.2.1.2-5. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Pearl Harbor

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.0	1.2×10^{-3}
MCW	N/A	N/A
NPA	7.0×10^{-1}	3.5×10^{-4}
MOI	1.8×10^{-1}	8.9×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	5.6×10^{2}	2.8×10^{-1}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2×10^{-3}
MCW	N/A	N/A
NPA	2.2×10^{1}	2.2×10^{-2}
MOI	6.7×10^{-1}	3.4×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	1.2×10^{3}	6.0×10^{-1}

Table F.1.4.2.1.2-6. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Norfolk

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.4 x 10^{-1}	2.9 x 10^{-4}
MCW	N/A	N/A
NPA	1.6 x 10^{-1}	8.2 x 10^{-5}
MOI	2.7 x 10^{-1}	1.3 x 10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	1.6 x 10^{2}	8.1 x 10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2 x 10^{-3}
MCW	N/A	N/A
NPA	1.8	8.8 x 10^{-4}
MOI	2.7	1.4 x 10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	7.0 x 10^{2}	3.5 x 10^{-1}

Table F.1.4.2.1.2-7. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Portsmouth

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.2 x 10^{-1}	2.9 x 10^{-4}
MCW	N/A	N/A
NPA	1.5 x 10^{-1}	7.7 x 10^{-5}
MOI	1.2 x 10^{-1}	5.9 x 10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	7.9 x 10^{1}	4.0 x 10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2 x 10^{-3}
MCW	N/A	N/A
NPA	3.3	1.6 x 10^{-3}
MOI	1.4	7.0 x 10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	2.9 x 10^{2}	1.5 x 10^{-1}

Table F.1.4.2.1.2-8. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Kesselring

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.4×10^{-1}	2.9×10^{-4}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	1.9×10^{-2}	9.7×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	5.6×10^{1}	2.8×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2×10^{-3}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	2.3×10^{-1}	1.2×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	2.2×10^{2}	1.1×10^{-1}

Table F.1.4.2.1.2-9. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Nevada Test Site

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.8×10^{-1}	1.9×10^{-4}
MCW	2.1×10^{-4}	8.0×10^{-8}
NPA	N/A	N/A
MOI	1.5×10^{-3}	7.3×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	4.3×10^{-1}	2.2×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2×10^{-3}
MCW	8.1×10^{-3}	3.3×10^{-6}
NPA	N/A	N/A
MOI	2.0×10^{-2}	9.9×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	1.4	7.0×10^{-4}

Table F.1.4.2.1.2-10. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Oak Ridge

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.0	1.2×10^{-3}
MCW	6.6×10^{-2}	2.6×10^{-5}
NPA	9.1×10^{-1}	4.6×10^{-4}
MOI	7.6×10^{-1}	3.8×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	7.4×10^{1}	3.7×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.0	3.2×10^{-3}
MCW	3.6×10^{-1}	1.4×10^{-4}
NPA	5.6	2.8×10^{-3}
MOI	4.7	2.4×10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	1.8×10^{2}	8.8×10^{-2}

F.1.4.2.1.3 Mechanical Damage from Operator Error, Crane Failure, or Similar Accidents

F.1.4.2.1.3.1 Description of Conditions. Accidental mechanical damage to spent fuel was evaluated. The hypothetical accident included damage to one fuel unit, allowing fission products within the elements to escape through the clad failures. All gas and some volatile and solid nuclides were calculated to be released to the pool. The release fractions are consistent with severe accident analyses and Regulatory Guide 1.4. Due to the presence of pool water, no solids would be released into the air inside the facility.

F.1.4.2.1.3.2 Source Term. Conditions used in developing the source term are as follows:

- One fuel unit is damaged because only one fuel unit would be handled at a time and the storage facility design prevents damage to stored units from such events.
- One percent of the fuel is damaged and those fission products are available for release.
- All (100%) of the noble gases are released to the environment.
- Approximately 25% of the halogens are released to the pool and 90% of these fission products are absorbed in the water as they rise through the pool water. Therefore, 2.5% of the halogens are released to the air inside the facility.
- Due to the gaseous nature of the released fission products, installed HEPA filters would not remove them once they are released to the air in the building.
- The release to the environment occurs at a constant rate over a 15-minute period.
- There is no particulate fission product release to the atmosphere due to the presence of pool water.

- The following amounts of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the possible exposure.

Nuclides	Curies
H-3	1.42
I-129	2.52×10^{-6}
I-131	5.37×10^{-5}

F.1.4.2.1.3.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical mechanical damage accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. The probability of the occurrence of fuel damage is small based on the conservative fuel handling rules. At the INEL-ECF, it is recognized that the drop of a heavy container into a storage rack could crush the rack and the stored fuel and so heavy casks are never moved over the storage rack area. The heavy containers are brought only into an empty receiving area to discharge a single fuel unit. The spent fuel is removed from the receiving area before the next fuel unit is brought into the receiving area. Therefore, two errors must occur before damaged fuel is possible. The first is that fuel is improperly left in the discharge station while the heavy cask is moved over the discharge station. The second is that the cask must accidentally fall from the overhead crane or the crane must fail. The probability of failure associated with crane failure has been taken as 10^{-2} per year. Further, the crane failure must also occur in the right location and the drop must be high enough that sufficient energy is available to damage both the discharge station structurals and the fuel inside. An additional factor of 10^{-2} has been taken for this event, giving the total probability of 10^{-4} for the drop of the cask in the right location. Allowing a fuel unit to remain in the stand requires an operator error because fuel handling procedures call for the fuel unit to be removed from the stand and taken to an underwater storage location away from the receiving area. In addition, because independent overchecking is required for all fuel movement, an error by a verifier is also required. Therefore, based on operator error rates (Swain and Guttman 1983), the likelihood of this error is taken as 10^{-4} per year. Hence, the combined probability of cask drop on a fuel unit is taken as 10^{-8} per year per fuel movement. Then, taking an estimated rate of 1,000 fuel movements per year, the overall probability is taken as 10^{-5} events per year.

Wet Storage Mechanical Damage Summary			
Site	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
INEL	2.6×10^{-6}	5.3×10^{-6}	5.3×10^{-11}
Savannah River	2.2×10^{-6}	2.0×10^{-5}	2.0×10^{-10}
Hanford	9.8×10^{-7}	8.6×10^{-6}	8.6×10^{-11}
Puget Sound	1.7×10^{-4}	7.2×10^{-5}	7.2×10^{-10}
Pearl Harbor	9.3×10^{-5}	1.5×10^{-4}	1.5×10^{-9}
Norfolk	3.5×10^{-4}	8.0×10^{-5}	8.0×10^{-10}
Portsmouth	1.9×10^{-4}	5.6×10^{-5}	5.6×10^{-10}
Kesselring	3.6×10^{-5}	6.0×10^{-5}	6.0×10^{-10}
Nevada Test Site	4.6×10^{-6}	5.6×10^{-7}	5.6×10^{-12}
Oak Ridge	5.9×10^{-4}	3.4×10^{-5}	3.4×10^{-10}

The risk for this hypothetical accident is generally more severe at Navy shipyards than at the DOE sites. At all sites, this accident results in the lowest or next to the lowest risk of the wet storage accidents evaluated.

For the hypothetical wet storage mechanical damage accident scenario, the radioactive plume might cause contamination of the ground to a downwind distance of less than 0.06 mile. This would yield a total area impacted by the accident of less than 0.5 acre. The calculated downwind distance would be contained within the boundaries of all sites under evaluation.

Table F.1.4.2.1.3-1. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At INEL

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.9×10^{-4}	7.6×10^{-8}
MCW	2.5×10^{-7}	9.6×10^{-11}
NPA	1.5×10^{-7}	7.4×10^{-11}
MOI	5.7×10^{-7}	2.9×10^{-10}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	5.0×10^{-3}	2.5×10^{-6}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	2.4×10^{-6}	9.6×10^{-10}
NPA	8.3×10^{-7}	4.2×10^{-10}
MOI	2.6×10^{-6}	1.3×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	1.1×10^{-2}	5.3×10^{-6}

Table F.1.4.2.1.3-2. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Savannah River

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.4×10^{-5}	3.4×10^{-8}
MCW	5.2×10^{-6}	2.1×10^{-9}
NPA	9.1×10^{-8}	4.5×10^{-11}
MOI (New ECF)	3.9×10^{-7}	1.9×10^{-10}
MOI (Barnwell)	1.5×10^{-6}	7.4×10^{-10}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	7.1×10^{-3}	3.5×10^{-6}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	6.7×10^{-5}	2.6×10^{-8}
NPA	1.4×10^{-6}	7.2×10^{-10}
MOI (New ECF)	2.2×10^{-6}	1.1×10^{-9}
MOI (Barnwell)	1.8×10^{-5}	9.0×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	4.1×10^{-2}	2.0×10^{-5}

Table F.1.4.2.1.3-3. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Hanford

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.4×10^{-5}	3.4×10^{-8}
MCW	7.1×10^{-6}	2.9×10^{-9}
NPA	1.0×10^{-7}	5.1×10^{-11}
MOI (New (ECF)	1.3×10^{-7}	6.5×10^{-11}
MOI (FMEF)	2.4×10^{-7}	1.2×10^{-10}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	9.4×10^{-4}	4.7×10^{-7}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	4.4×10^{-5}	1.8×10^{-8}
NPA	1.6×10^{-6}	7.9×10^{-10}
MOI (New ECF)	9.8×10^{-7}	4.9×10^{-10}
MOI (FMEF)	3.1×10^{-6}	1.5×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	1.7×10^{-2}	8.6×10^{-6}

Table F.1.4.2.1.3-4. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Puget Sound

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.6×10^{-5}	1.8×10^{-8}
MCW	N/A	N/A
NPA	5.5×10^{-5}	2.7×10^{-8}
MOI	1.3×10^{-5}	6.7×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	6.0×10^{-3}	3.0×10^{-6}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	N/A	N/A
NPA	6.5×10^{-4}	3.2×10^{-7}
MOI	1.7×10^{-4}	8.4×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	1.5×10^{-1}	7.2×10^{-5}

Table F.1.4.2.1.3-5. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Pearl Harbor

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.9×10^{-4}	7.6×10^{-8}
MCW	N/A	N/A
NPA	4.9×10^{-5}	2.4×10^{-8}
MOI	2.3×10^{-5}	1.2×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	1.1×10^{-1}	5.6×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	N/A	N/A
NPA	1.6×10^{-3}	7.9×10^{-7}
MOI	9.3×10^{-5}	4.6×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	3.1×10^{-1}	1.5×10^{-4}

Table F.1.4.2.1.3-6. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Norfolk

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.6×10^{-5}	1.9×10^{-8}
MCW	N/A	N/A
NPA	1.2×10^{-5}	6.0×10^{-9}
MOI	3.2×10^{-5}	1.6×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	1.4×10^{-2}	7.0×10^{-6}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	N/A	N/A
NPA	1.4×10^{-4}	7.0×10^{-8}
MOI	3.5×10^{-4}	1.7×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	1.6×10^{-1}	8.0×10^{-5}

Table F.1.4.2.1.3-7. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Portsmouth

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.6×10^{-5}	1.8×10^{-8}
MCW	N/A	N/A
NPA	1.1×10^{-5}	5.6×10^{-9}
MOI	1.5×10^{-5}	7.4×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	3.8×10^{-3}	1.9×10^{-6}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	N/A	N/A
NPA	2.5×10^{-4}	1.3×10^{-7}
MOI	1.9×10^{-4}	9.3×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	1.1×10^{-1}	5.6×10^{-5}

Table F.1.4.2.1.3-8. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Kesselring

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.6×10^{-5}	1.9×10^{-8}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	3.2×10^{-6}	1.6×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	4.7×10^{-2}	2.3×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	3.6×10^{-5}	1.8×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	1.2×10^{-1}	6.0×10^{-5}

Table F.1.4.2.1.3-9. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Nevada Test Site

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.0×10^{-5}	1.2×10^{-8}
MCW	3.0×10^{-8}	1.5×10^{-11}
NPA	N/A	N/A
MOI	3.8×10^{-7}	1.9×10^{-10}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	4.5×10^{-4}	2.3×10^{-7}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	1.8×10^{-6}	7.1×10^{-10}
NPA	N/A	N/A
MOI	4.6×10^{-6}	2.3×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	1.1×10^{-3}	5.6×10^{-7}

Table F.1.4.2.1.3-10. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Oak Ridge

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.9×10^{-4}	7.6×10^{-8}
MCW	5.4×10^{-6}	2.2×10^{-9}
NPA	6.6×10^{-5}	3.3×10^{-8}
MOI	9.3×10^{-5}	4.7×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	2.0×10^{-2}	1.0×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.2×10^{-4}	2.1×10^{-7}
MCW	3.3×10^{-5}	1.3×10^{-8}
NPA	4.2×10^{-4}	2.1×10^{-7}
MOI	5.9×10^{-4}	3.0×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	6.7×10^{-2}	3.4×10^{-5}

F.1.4.2.1.4 Airplane Crash.

F.1.4.2.1.4.1 Description of Conditions. Impact into water pools by aircraft with resulting damage to the naval fuel units stored inside the pool was evaluated. Based on the probability of occurrence, as discussed in Section F.3, specific analyses were only performed for Savannah River, the Nevada Test Site, Oak Ridge, Pearl Harbor, Norfolk, and Kesselring locations. At other locations, the likelihood of occurrence is less than 10^{-7} per year. The hypothetical accident included damage to all fuel units stored at the water pool. Fission products and corrosion products are released from the fuel units into the water pool; however, the pool water is not released to the environment. An airplane crash into a water pool would not produce enough force to cause the pool to leak because the walls of the water pool are constructed of thick, reinforced concrete with earth surrounding them, making them very strong. In addition, it was judged unlikely that an airplane would impact the water pool at an angle steep enough to expose the floor of the pool or the walls of the pool below the water level to the direct impact. The presence of pool water results in only a release of gaseous fission products to the atmosphere.

F.1.4.2.1.4.2 Source Term. Conditions used in developing the source term are as follows:

- One percent of the fission products from each of the fuel units stored inside the pool is available for release.

- Of the available fission products, 100% of the noble gases and 25% of the halogens are released to the pool water. Due to the presence of pool water, a reduction of the halogen release by a factor of 10 prior to release to the atmosphere occurs.

- No solid fission products or corrosion products are released to the environment due to the continued presence of pool water.

- The release to the environment occurs at a constant rate over a 15-minute period.

- 300 naval fuel units would be in the water pool.

- No filtration by HEPA filters is assumed.

- The following amounts of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the possible exposure.

Nuclide	Curies
I-129	7.59×10^{-4}
I-131	1.61×10^{-2}
H-3	4.28×10^{2}

F.1.4.2.1.4.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical airplane crash accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence.

Water Pool Airplane Crash Summary				
Site	Probability of accident per year	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
Savannah River	2×10^{-6}	6.4×10^{-4}	6.1×10^{-3}	1.2×10^{-8}
Pearl Harbor	2×10^{-5}	2.8×10^{-2}	4.6×10^{-2}	9.2×10^{-7}
Norfolk	4×10^{-7}	1.1×10^{-1}	2.4×10^{-2}	9.6×10^{-9}
Kesselring	2×10^{-7}	1.1×10^{-2}	1.8×10^{-2}	3.6×10^{-9}
Nevada Test Site	4×10^{-7}	1.3×10^{-3}	1.7×10^{-4}	6.8×10^{-11}
Oak Ridge	1×10^{-6}	1.8×10^{-1}	1.0×10^{-2}	1.0×10^{-8}

The risk for this hypothetical accident is most severe at Pearl Harbor. For the sites with crash probabilities less than 10^{-7} per year, consequences were not calculated since it is expected that they would not substantially contribute to the risk.

For the hypothetical airplane crash into a wet storage facility accident scenario, the radioactive plume might result in contamination of the ground to a downwind distance of less than 0.06 mile. This would yield a total area impacted by the accident of less than 0.5 acre. The calculated downwind distance would be contained within the boundaries of all sites that are at risk for this accident.

Table F.1.4.2.1.4-1. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Savannah River

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.5×10^{-2}	1.0×10^{-5}
MCW	1.6×10^{-3}	6.3×10^{-7}
NPA	2.8×10^{-5}	1.4×10^{-8}
MOI	1.1×10^{-4}	5.5×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	2.2	1.1×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{-1}	6.3×10^{-5}
MCW	2.0×10^{-2}	8.0×10^{-6}
NPA	4.3×10^{-4}	2.2×10^{-7}
MOI	6.4×10^{-4}	3.2×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	1.2×10^{1}	6.1×10^{-3}

Table F.1.4.2.1.4-2. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Pearl Harbor

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.7×10^{-2}	2.3×10^{-5}
MCW	N/A	N/A
NPA	1.5×10^{-2}	7.3×10^{-6}
MOI	6.9×10^{-3}	3.5×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	3.3×10^{1}	1.7×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{-1}	6.3×10^{-5}
MCW	N/A	N/A
NPA	4.7×10^{-1}	2.4×10^{-4}
MOI	2.8×10^{-2}	1.4×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	9.2×10^{1}	4.6×10^{-2}

Table F.1.4.2.1.4-3. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Norfolk

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.4×10^{-2}	5.6×10^{-6}
MCW	N/A	N/A
NPA	3.6×10^{-3}	1.8×10^{-6}
MOI	9.6×10^{-3}	4.8×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	4.2	2.1×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{-1}	6.3×10^{-5}
MCW	N/A	N/A
NPA	4.2×10^{-2}	2.1×10^{-5}
MOI	1.1×10^{-1}	5.3×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	4.8×10^{1}	2.4×10^{-2}

Table F.1.4.2.1.4-4. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Kesselring

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.4×10^{-2}	5.6×10^{-6}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	9.5×10^{-4}	4.8×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	1.4×10^{1}	7.1×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{-1}	6.3×10^{-5}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	1.1×10^{-2}	5.4×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	3.6×10^{1}	1.8×10^{-2}

Table F.1.4.2.1.4-5. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Nevada Test Site

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.0×10^{-3}	3.6×10^{-6}
MCW	9.1×10^{-6}	3.7×10^{-9}
NPA	N/A	N/A
MOI	5.5×10^{-5}	2.8×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	1.3×10^{-1}	6.5×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{-1}	6.4×10^{-5}
MCW	5.3×10^{-4}	2.2×10^{-7}
NPA	N/A	N/A
MOI	1.3×10^{-3}	6.5×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	3.3×10^{-1}	1.7×10^{-4}

Table F.1.4.2.1.4-6. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Oak Ridge

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.7×10^{-2}	2.3×10^{-5}
MCW	1.6×10^{-3}	6.5×10^{-7}
NPA	2.0×10^{-2}	9.9×10^{-6}
MOI	2.8×10^{-2}	1.4×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	6.0	3.0×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{-1}	6.3×10^{-5}
MCW	9.9×10^{-3}	3.9×10^{-6}
NPA	1.3×10^{-1}	6.3×10^{-5}
MOI	1.8×10^{-1}	8.9×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	2.0×10^{1}	1.0×10^{-2}

F.1.4.2.1.5 HEPA Filter Fire.

F.1.4.2.1.5.1 Description of Conditions. In this hypothetical accident scenario, a fire in the ECF High Efficiency Particulate Air (HEPA) filter banks is postulated. This accident could be initiated by the ignition of a flammable mixture released upstream of the system or by an external, unrelated fire that spreads to this system. Although the risks associated with this accident are relatively minor, it was analyzed to bound the higher probability, lower consequence type accident category. The airborne release fractions associated with this accident were conservatively chosen so that a HEPA filter failure by crushing or impact was also bounded.

F.1.4.2.1.5.2 Source Term. Conditions used in developing the source term are as follows:

- The original inventory of fission products in the filters is based on the total estimated unabated ECF releases over a 5-year period.
- One percent of the radionuclide inventory present on the filters becomes airborne during the fire. Release fractions for HEPA filters are small because the filters are constructed of material containing glass fibers which would melt during a fire and trap particles in the medium. Measurements from experiments show that one one-hundredth of 1% of the material in HEPA filters could be released during a fire, but 1% has been used in these analyses to allow for uncertainties in the final results of an individual fire.
- The release to the environment occurs at a constant rate over a 15-minute period.
- There is no increase in direct radiation due to this accident.
- The following amounts of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the possible exposure.
- No filtration by HEPA filters is assumed.

Nuclide	Curies	Nuclide	Curies
Cs-137	1.46×10^{-3}	Co-60	2.09×10^{-3}
Cs-134	2.04×10^{-4}	Sr-90	8.90×10^{-4}
Ba-137M	6.26×10^{-6}	Y-90	8.90×10^{-4}
Fe-55	2.32×10^{-3}	Eu-154	9.80×10^{-5}
Ni-63	2.98×10^{-3}		

F.1.4.2.1.5.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical HEPA filter fire accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. The probability of a fire in a HEPA filter is estimated based on the probability of other fires spreading to the HEPA filter system. As discussed in section F.2.4.2, a probability of 5×10^{-3} is assigned to chemical fires. The probability of HEPA fires is considered less than a chemical fire since chemicals would not be stored in the immediate vicinity of the HEPA filter system. Additionally, HEPA filters are not inherently volatile or explosive. It is estimated that the probability for an existing chemical fire to spread to the HEPA filters is less than 0.1. This results in a probability of less than 5×10^{-4} for a HEPA filter fire. A value of 5×10^{-4} was used to develop the risk results in the table.

HEPA Filter Fire Summary			
Site	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
INEL	2.5×10^{-5}	5.3×10^{-5}	2.7×10^{-8}
Savannah River	2.1×10^{-5}	1.3×10^{-4}	6.5×10^{-8}
Hanford	7.0×10^{-6}	5.3×10^{-5}	2.7×10^{-8}
Puget Sound	1.6×10^{-3}	6.4×10^{-4}	3.2×10^{-7}
Pearl Harbor	8.7×10^{-4}	1.2×10^{-3}	6.0×10^{-7}
Norfolk	3.3×10^{-3}	6.9×10^{-4}	3.5×10^{-7}
Portsmouth	1.7×10^{-3}	3.9×10^{-4}	2.0×10^{-7}
Kesselring	3.5×10^{-4}	3.3×10^{-4}	1.7×10^{-7}
Nevada Test Site	4.3×10^{-5}	5.7×10^{-6}	2.9×10^{-9}
Oak Ridge	5.7×10^{-3}	2.2×10^{-4}	1.1×10^{-7}

The risk for this hypothetical accident is generally more severe at the Navy shipyards than at the DOE sites.

For the hypothetical HEPA filter fire accident scenario, the radioactive plume might cause contamination of the ground to a downwind distance of less than 0.06 mile. This would yield a total area impacted by the accident of less than 0.5 acre. The calculated downwind distance would be contained within the boundaries of all sites under evaluation.

Table F.1.4.2.1.5-1. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At INEL

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.7×10^{-4}	3.5×10^{-7}
MCW	7.9×10^{-7}	3.2×10^{-10}
NPA	4.5×10^{-7}	2.2×10^{-10}
MOI	9.9×10^{-6}	5.0×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	7.6×10^{-2}	3.8×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	8.8×10^{-6}	3.5×10^{-9}
NPA	2.7×10^{-6}	1.4×10^{-9}
MOI	2.5×10^{-5}	1.3×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	1.1×10^{-1}	5.3×10^{-5}

Table F.1.4.2.1.5-2. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Savannah River

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.9×10^{-4}	1.5×10^{-7}
MCW	2.3×10^{-5}	8.8×10^{-9}
NPA	2.9×10^{-7}	1.4×10^{-10}
MOI (New ECF)	7.2×10^{-6}	3.6×10^{-9}
MOI (Barnwell)	1.7×10^{-5}	8.6×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	4.1×10^{-2}	2.0×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	2.9×10^{-4}	1.1×10^{-7}
NPA	4.9×10^{-6}	2.5×10^{-9}
MOI (New ECF)	2.1×10^{-5}	1.0×10^{-8}
MOI (Barnwell)	1.6×10^{-4}	8.1×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	2.5×10^{-1}	1.3×10^{-4}

Table F.1.4.2.1.5-3. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Hanford

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.9×10^{-4}	1.5×10^{-7}
MCW	3.0×10^{-5}	1.2×10^{-8}
NPA	3.5×10^{-7}	1.8×10^{-10}
MOI (New ECF)	9.6×10^{-7}	4.8×10^{-10}
MOI (FMEF)	1.9×10^{-6}	9.7×10^{-10}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	6.7×10^{-3}	3.4×10^{-6}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	1.9×10^{-4}	7.5×10^{-8}
NPA	5.5×10^{-6}	2.7×10^{-9}
MOI (New ECF)	7.0×10^{-6}	3.5×10^{-9}
MOI (FMEF)	2.4×10^{-5}	1.2×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	1.1×10^{-1}	5.3×10^{-5}

Table F.1.4.2.1.5-4. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Puget Sound

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1×10^{-4}	8.4×10^{-8}
MCW	N/A	N/A
NPA	2.5×10^{-4}	1.2×10^{-7}
MOI	1.4×10^{-4}	6.8×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	3.4×10^{-1}	1.7×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	N/A	N/A
NPA	2.9×10^{-3}	1.5×10^{-6}
MOI	1.6×10^{-3}	8.0×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	1.3	6.4×10^{-4}

Table F.1.4.2.1.5-5. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Pearl Harbor

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.7×10^{-4}	3.5×10^{-7}
MCW	N/A	N/A
NPA	2.2×10^{-4}	1.1×10^{-7}
MOI	2.2×10^{-4}	1.1×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	9.0×10^{-1}	4.5×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	N/A	N/A
NPA	7.2×10^{-3}	3.6×10^{-6}
MOI	8.7×10^{-4}	4.3×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	2.4	1.2×10^{-3}

Table F.1.4.2.1.5-6. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Norfolk

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1×10^{-4}	8.5×10^{-8}
MCW	N/A	N/A
NPA	5.3×10^{-5}	2.7×10^{-8}
MOI	3.2×10^{-4}	1.6×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	2.3×10^{-1}	1.2×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	N/A	N/A
NPA	6.2×10^{-4}	3.1×10^{-7}
MOI	3.3×10^{-3}	1.7×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	1.4	6.9×10^{-4}

Table F.1.4.2.1.5-7. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Portsmouth

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1×10^{-4}	8.4×10^{-8}
MCW	N/A	N/A
NPA	5.0×10^{-5}	2.5×10^{-8}
MOI	1.4×10^{-4}	7.2×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	1.2×10^{-1}	6.0×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	N/A	N/A
NPA	1.1×10^{-3}	5.6×10^{-7}
MOI	1.7×10^{-3}	8.7×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	7.9×10^{-1}	3.9×10^{-4}

Table F.1.4.2.1.5-8. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Kesselring

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.1×10^{-4}	8.5×10^{-8}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	5.5×10^{-5}	2.7×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	2.0×10^{-1}	9.8×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	3.5×10^{-4}	1.8×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	6.7×10^{-1}	3.3×10^{-4}

Table F.1.4.2.1.5-9. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Nevada Test Site

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.4×10^{-4}	5.5×10^{-8}
MCW	1.1×10^{-7}	4.2×10^{-11}
NPA	N/A	N/A
MOI	8.5×10^{-6}	4.2×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	7.6×10^{-3}	3.8×10^{-6}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	6.2×10^{-6}	2.5×10^{-9}
NPA	N/A	N/A
MOI	4.3×10^{-5}	2.2×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	1.1×10^{-2}	5.7×10^{-6}

Table F.1.4.2.1.5-10. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Oak Ridge

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.7×10^{-4}	3.5×10^{-7}
MCW	2.3×10^{-5}	8.8×10^{-9}
NPA	3.0×10^{-4}	1.5×10^{-7}
MOI	9.0×10^{-4}	4.5×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	1.2×10^{-1}	6.0×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.4×10^{-3}	9.6×10^{-7}
MCW	1.4×10^{-4}	5.6×10^{-8}
NPA	1.9×10^{-3}	9.4×10^{-7}
MOI	5.7×10^{-3}	2.9×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	4.3×10^{-1}	2.2×10^{-4}

F.1.4.2.1.6 Minor Water Pool Leakage.

F.1.4.2.1.6.1 Description of Conditions. In this hypothetical accident scenario, a minor leak develops in the water pool resulting in a gradual discharge to the environment. There is no danger of uncovering any spent nuclear fuel in the water pool, since the leak is so small that it is undetected and water level is maintained in the water pool. Since a strict accounting of water added to and removed from the water pool is maintained, the magnitude of this leak would be less than 4,400 gallons per year. The 4,400 gallons per year value is the maximum amount of water which might leak out of the water pool before periodic review of the water balance would detect a leak.

F.1.4.2.1.6.2 Source Term. There is no airborne release above normal levels in this hypothetical accident scenario. The radionuclide inventory in the leaking water is based on radioactivity analysis of ECF water pool water. The isotopes that were analyzed for but not detected could exist at the minimum detection limit.

Nuclide	Sample Results (μCi/mL)	10CFR20 Effluent Limit (μCi/mL)	Annual Releases (Ci/year)
H-3	2.0×10^{-4}	1.0×10^{-3}	3.3×10^{-3}
Mn-54	2.5×10^{-8}	3.0×10^{-5}	4.1×10^{-7}
Fe-55	1.0×10^{-8} *	1.0×10^{-4}	1.6×10^{-7} *
Co-58	7.0×10^{-8}	2.0×10^{-5}	1.1×10^{-6}
Co-60	1.6×10^{-5}	3.0×10^{-6}	2.6×10^{-5}
Ni-63	2.3×10^{-7}	1.0×10^{-4}	3.8×10^{-6}
Sr-90	4.0×10^{-9}	5.0×10^{-7}	6.5×10^{-8}
Y-90	4.0×10^{-9}	7.0×10^{-6}	6.5×10^{-8}
I-129	4.0×10^{-7} *	2.0×10^{-7}	6.5×10^{-6} *
Cs-137	4.2×10^{-8}	1.0×10^{-6}	6.9×10^{-7}

* These radionuclides were not detected in the ECF water. The numbers quoted reflect the detection limit of the analysis.

It should be noted that the sample results for the water pool indicate that the nuclide levels are all below the Code of Federal Regulations limits for liquid effluent in 10CFR20 with the exception of Co-60. The level of I-129 used in the calculations was based on the minimum detection limit of the sample. This level exceeds the effluent limit; however, I-129 was not actually detected in the water sample. Since Sr-90 has comparable water solubility to I-129 and exists in spent nuclear fuel at about a factor of 1.0 x 10^6 higher than I-129, it is inferred from the detected level of Sr-90 that the actual level of I-129 is well below the 10CFR20 effluent limit.

F.1.4.2.1.6.3 Results. The following table summarizes the public health risk to the general population that might result from the hypothetical minor water pool leak at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. The probability of a leak developing is 10^{-1} per year.

Minor Water Pool Leakage Summary			
Site	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
INEL	2.5 x 10^{-9}	1.3 x 10^{-8}	1.3 x 10^{-9}
Savannah River	7.9 x 10^{-10}	1.3 x 10^{-9}	1.3 x 10^{-10}
Hanford	9.9 x 10^{-12}	1.7 x 10^{-10}	1.7 x 10^{-11}
Puget Sound	3.2 x 10^{-10}	4.2 x 10^{-9}	4.2 x 10^{-10}
Pearl Harbor	1.3 x 10^{-10}	4.6 x 10^{-10}	4.6 x 10^{-11}
Norfolk	2.7 x 10^{-10}	1.8 x 10^{-9}	1.8 x 10^{-10}
Portsmouth	1.3 x 10^{-10}	1.4 x 10^{-9}	1.4 x 10^{-10}
Kesselring	6.0 x 10^{-9}	8.5 x 10^{-9}	8.5 x 10^{-10}
Nevada Test Site	2.5 x 10^{-9}	1.4 x 10^{-9}	1.4 x 10^{-10}
Oak Ridge	1.5 x 10^{-9}	3.9 x 10^{-9}	3.9 x 10^{-10}

At all sites except the Nevada Test Site, this accident results in the lowest or next to lowest risk of the wet storage accidents evaluated.

Table F.1.4.2.1.6-1. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At INEL

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	1.6 x 10^{-13}	6.4 x 10^{-17}
NPA	1.6 x 10^{-13}	8.0 x 10^{-17}
MOI	2.5 x 10^{-9}	1.3 x 10^{-12}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	2.6 x 10^{-5}	1.3 x 10^{-8}

Table F.1.4.2.1.6-2. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Savannah River

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	4.8 x 10^{-13}	1.9 x 10^{-16}
NPA	4.8 x 10^{-13}	2.4 x 10^{-16}
MOI	7.9 x 10^{-10}	4.0 x 10^{-13}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	2.5 x 10^{-6}	1.3 x 10^{-9}

Table F.1.4.2.1.6-3. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Hanford

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	8.3×10^{-15}	3.3×10^{-18}
NPA	8.3×10^{-15}	4.2×10^{-18}
MOI	9.9×10^{-12}	5.0×10^{-15}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	3.3×10^{-7}	1.7×10^{-10}

Table F.1.4.2.1.6-4. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Puget Sound

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	N/A	N/A
NPA	1.2×10^{-11}	6.0×10^{-15}
MOI	3.2×10^{-10}	1.6×10^{-13}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	8.4×10^{-6}	4.2×10^{-9}

Table F.1.4.2.1.6-5. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Pearl Harbor

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	N/A	N/A
NPA	4.8×10^{-12}	2.4×10^{-15}
MOI	1.3×10^{-10}	6.5×10^{-14}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	9.2×10^{-7}	4.6×10^{-10}

Table F.1.4.2.1.6-6. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Norfolk

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	N/A	N/A
NPA	9.9×10^{-12}	5.0×10^{-15}
MOI	2.7×10^{-10}	1.4×10^{-13}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	3.6×10^{-6}	1.8×10^{-9}

Table F.1.4.2.1.6-7. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Portsmouth

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	N/A	N/A
NPA	4.8×10^{-12}	2.4×10^{-15}
MOI	1.3×10^{-10}	6.5×10^{-14}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	2.7×10^{-6}	1.4×10^{-9}

Table F.1.4.2.1.6-8. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Kesselring

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	N/A	N/A
NPA	N/A	N/A
MOI	6.0×10^{-9}	3.0×10^{-12}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	1.7×10^{-5}	8.5×10^{-9}

Table F.1.4.2.1.6-9. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Nevada Test Site

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	1.6×10^{-13}	6.4×10^{-17}
NPA	1.6×10^{-13}	8.0×10^{-17}
MOI	2.5×10^{-9}	1.3×10^{-12}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	2.7×10^{-6}	1.4×10^{-9}

Table F.1.4.2.1.6-10. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Oak Ridge

Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	N/A	N/A
MCW	9.4×10^{-13}	3.8×10^{-16}
NPA	9.4×10^{-13}	4.7×10^{-16}
MOI	1.5×10^{-9}	7.5×10^{-13}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	7.7×10^{-6}	3.9×10^{-9}

F.1.4.2.2 Dry Storage.

F.1.4.2.2.1 Wind-driven Missile Impact into Storage Casks with Mechanical Damage.

F.1.4.2.2.1.1 Description of Conditions. In this hypothetical accident, no fuel damage would result from any impact because of the strength of the containers used. Dry storage containers could experience a major wind storm or tornado which could propel a large object into a storage container causing the container seal to be breached. However, container analysis for this situation shows that the container is strong enough to prevent crushing of the spent nuclear fuel and release of fission products.

Winds produced by tornados are higher than hurricane winds and thus the impacting missile would be travelling with higher velocity and would have higher kinetic energy. Even at this higher velocity, analysis has shown that the missile would not penetrate the container. The probability of penetration at the lower velocity of a hurricane (212 miles per hour) would be even smaller than the probability of penetration for a missile propelled by the winds of a tornado (travelling at 360 mph). While hurricanes can have high winds, hurricane winds normally cannot generate the very large, very fast missiles analyzed for tornados. While hurricanes may occur more frequently than tornados, the overall risk from a hurricane is lower because the container would not be penetrated.

The analysis of wind damage using missiles propelled by the winds of tornados is the same as is done for design of nuclear power plants. Hurricanes very infrequently have winds that could generate such missiles, so the analyses provided for tornados provide an upper limit for the effects of hurricanes. Examination of damage caused by recent severe hurricanes shows that robust structures can withstand hurricanes.

F.1.4.2.2.1.2 Source Term. Conditions used in developing the source term are as follows:

- The source term is based on best estimate spent nuclear fuel corrosion products.
- One percent of the original corrosion products associated with the fuel could be released from the cask to the atmosphere. This is based on experimental

measurements of the fraction of corrosion products loosened from naval spent nuclear fuel by shock and vibration and the fact that a wind-driven missile would not penetrate the container or damage the fuel inside. Only loose corrosion products would be available for release from the container, and any release from the container would have to occur via a convoluted path through the damaged seal.

- The release to the environment occurs at a constant rate over a 15-minute period.
- There is no increase in direct radiation due to this accident.
- The following amounts of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the possible exposure.

Nuclide	Curies
Co-60	9.58×10^{-2}
Fe-55	1.76×10^{-1}
Co-58	3.54×10^{-2}
Mn-54	5.98×10^{-3}
Fe-59	5.11×10^{-4}

F.1.4.2.2.1.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical wind-driven missile accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. The probability of container damage is small due to the very strong container design. The dry storage containers are expected to be designed as well as shipping containers so that they would not be penetrated by environmentally caused missiles and the fuel would not be affected. However, an analysis was performed for a case in which the impact of a tornado missile might topple a container on a railcar and cause unseating of the container seal and thus release radioactive material in the form of corrosion products.

The probability of the occurrence of a tornado was obtained using the data in document WASH-1300 (AEC 1974). The maximum likelihood of a tornado occurrence at all storage locations being evaluated in the continental United States is 10^{-3} per year. The probability of a missile generated by the tornado striking a container and causing the damage analyzed has been estimated to be less than 10^{-2}. Thus, the total probability of a wind-driven missile damaging a container is less than 10^{-5}, and a probability of 10^{-5} per year was used in the risk assessment.

Dry Storage Mechanical Damage Summary			
Site	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
INEL	4.6×10^{-4}	4.9×10^{-4}	4.9×10^{-9}
Savannah River	4.9×10^{-4}	3.0×10^{-3}	3.0×10^{-8}
Hanford	1.7×10^{-4}	1.3×10^{-3}	1.3×10^{-8}
Puget Sound	3.9×10^{-2}	1.7×10^{-2}	1.7×10^{-7}
Pearl Harbor	2.1×10^{-2}	3.0×10^{-2}	3.0×10^{-7}
Norfolk	8.1×10^{-2}	1.8×10^{-2}	1.8×10^{-7}
Portsmouth	4.2×10^{-2}	1.0×10^{-2}	1.0×10^{-7}
Kesselring	8.1×10^{-3}	7.4×10^{-3}	7.4×10^{-8}
Nevada Test Site	8.8×10^{-4}	5.3×10^{-5}	5.3×10^{-10}
Oak Ridge	1.4×10^{-1}	5.1×10^{-3}	5.1×10^{-8}

The risk for this hypothetical accident is generally more severe at Navy shipyards than at the DOE sites. This accident results in the lowest risk of the two dry storage accidents evaluated.

For the hypothetical wind-driven missile accident scenario, the radioactive plume might cause contamination of the ground to a downwind distance of less than 0.06 mile. This would yield a total area impacted by the accident of less than 0.5 acre. The calculated downwind distance would be contained within the boundaries of all sites under evaluation.

Table F.1.4.2.2.1-1. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At INEL

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.0×10^{-2}	8.0×10^{-6}
MCW	1.8×10^{-5}	9.2×10^{-9}
NPA	1.0×10^{-5}	5.2×10^{-9}
MOI	8.0×10^{-5}	4.0×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	2.3×10^{-1}	1.2×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	2.0×10^{-4}	1.0×10^{-7}
NPA	6.3×10^{-5}	3.1×10^{-8}
MOI	4.6×10^{-4}	2.3×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	9.8×10^{-1}	4.9×10^{-4}

Table F.1.4.2.2.1-2. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Savannah River

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.9×10^{-3}	3.6×10^{-6}
MCW	5.3×10^{-4}	2.1×10^{-7}
NPA	6.7×10^{-6}	3.4×10^{-9}
MOI (New ECF)	1.6×10^{-4}	8.1×10^{-8}
MOI (Barnwell)	4.0×10^{-4}	2.0×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	9.4×10^{-1}	4.7×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	6.7×10^{-3}	2.6×10^{-6}
NPA	1.1×10^{-4}	5.7×10^{-8}
MOI (New ECF)	4.9×10^{-4}	2.5×10^{-7}
MOI (Barnwell)	3.9×10^{-3}	2.0×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	6.1	3.0×10^{-3}

Table F.1.4.2.2.1-3. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Hanford

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.9×10^{-3}	3.6×10^{-6}
MCW	7.0×10^{-4}	2.8×10^{-7}
NPA	8.1×10^{-6}	4.1×10^{-9}
MOI (New ECF)	2.3×10^{-5}	1.1×10^{-8}
MOI (FMEF)	4.6×10^{-5}	2.3×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	1.4×10^{-1}	7.0×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	4.4×10^{-3}	1.8×10^{-6}
NPA	1.3×10^{-4}	6.3×10^{-8}
MOI (New ECF)	1.7×10^{-4}	8.4×10^{-8}
MOI (FMEF)	5.9×10^{-4}	2.9×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	2.5	1.3×10^{-3}

Table F.1.4.2.2.1-4. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Puget Sound

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.9×10^{-3}	1.9×10^{-6}
MCW	N/A	N/A
NPA	5.7×10^{-3}	2.9×10^{-6}
MOI	3.5×10^{-3}	1.7×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	1.2×10^{1}	5.8×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	N/A	N/A
NPA	6.8×10^{-2}	3.4×10^{-5}
MOI	3.9×10^{-2}	1.9×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2975810	3.4×10^{1}	1.7×10^{-2}

Table F.1.4.2.2.1-5. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Pearl Harbor

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.0×10^{-2}	8.0×10^{-6}
MCW	N/A	N/A
NPA	5.2×10^{-3}	2.6×10^{-6}
MOI	5.3×10^{-3}	2.7×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	2.2×10^{1}	1.1×10^{-2}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	N/A	N/A
NPA	1.7×10^{-1}	8.4×10^{-5}
MOI	2.1×10^{-2}	1.1×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	5.9×10^{1}	3.0×10^{-2}

Table F.1.4.2.2.1-6. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Norfolk

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.9×10^{-3}	2.0×10^{-6}
MCW	N/A	N/A
NPA	1.2×10^{-3}	6.2×10^{-7}
MOI	7.8×10^{-3}	3.9×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	7.4	3.7×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	N/A	N/A
NPA	1.4×10^{-2}	7.1×10^{-6}
MOI	8.1×10^{-2}	4.0×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	3.5×10^{1}	1.8×10^{-2}

Table F.1.4.2.2.1-7. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Portsmouth

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.9×10^{-3}	1.9×10^{-6}
MCW	N/A	N/A
NPA	1.2×10^{-3}	5.8×10^{-7}
MOI	3.5×10^{-3}	1.8×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	4.2	2.1×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	N/A	N/A
NPA	2.6×10^{-2}	1.3×10^{-5}
MOI	4.2×10^{-2}	2.1×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	2.0×10^{1}	1.0×10^{-2}

Table F.1.4.2.2.1-8. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Kesselring

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	4.9×10^{-3}	2.0×10^{-6}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	8.8×10^{-4}	4.4×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	3.3	1.7×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	8.1×10^{-3}	4.0×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	1.5×10^{1}	7.4×10^{-3}

Table F.1.4.2.2.1-9. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Nevada Test Site

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.2×10^{-3}	1.3×10^{-6}
MCW	2.5×10^{-6}	9.6×10^{-10}
NPA	N/A	N/A
MOI	4.5×10^{-5}	2.2×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	1.5×10^{-2}	7.3×10^{-6}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	1.4×10^{-4}	5.8×10^{-8}
NPA	N/A	N/A
MOI	8.8×10^{-4}	4.4×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	1.1×10^{-1}	5.3×10^{-5}

Table F.1.4.2.2.1-10. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Oak Ridge

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.0×10^{-2}	8.0×10^{-6}
MCW	5.3×10^{-4}	2.1×10^{-7}
NPA	6.9×10^{-3}	3.4×10^{-6}
MOI	2.2×10^{-2}	1.1×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	2.8	1.4×10^{-3}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.6×10^{-2}	2.2×10^{-5}
MCW	3.2×10^{-3}	1.3×10^{-6}
NPA	4.4×10^{-2}	2.2×10^{-5}
MOI	1.4×10^{-1}	6.9×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	1.0×10^{1}	5.1×10^{-3}

F.1.4.2.2.2 Airplane Crash.

F.1.4.2.2.2.1 Description of Conditions. A hypothetical aircraft accident scenario was developed for the dry storage option. Based on the probability of occurrence, as discussed in Section F.3, specific analyses were only performed for Savannah River, Oak Ridge, Pearl Harbor, Norfolk, Portsmouth, and Kesselring locations. At other locations, the likelihood of occurrence is less than 10^{-7} per year. The accident is postulated to cause damage to a single storage cask. This is based on the fact that containers used to store naval spent nuclear fuel would be very rugged so that only the rotor shaft from one of an airliner's jet engines would be strong enough and possess enough energy to have a chance of penetrating a container. From analyses of existing container designs, the rotor of a large jet engine, including those from the largest aircraft such as a Boeing 777, Russian Antonov An-225, or a Lockheed C-5, would not penetrate a container during an airliner crash, but, for the purposes of evaluation, calculations were performed for one container damaged to the extent that fission products and corrosion products might be released. Due to the severity of the shock, the cask seal might be breached resulting in damage to the fuel. The severe mechanical shock results in the release of corrosion products to the environment. The release of fission products also occurs due to the impact and resultant fire. The fission product release factors are based on overheating testing performed on the naval fuel systems.

F.1.4.2.2.2.2 Source Term. Conditions used in developing the source term are as follows:

- One percent of all of the fuel units stored inside the cask are damaged either by the impact or the resultant fire and those fission products are available for release.
- Of the available fission products, 100% of the noble gases, 3% of the halogens, 1.1% of the cesium, and 0.1% of the remaining solids are released to the environment.
- The release to the environment occurs at a constant rate over a 15-minute period.
- Ten percent of the original corrosion products from the fuel units are released from the cask to the atmosphere.

- The following amount of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the possible exposure.

Nuclide	Curies
Cs-134	2.57×10^{1}
Cs-137	3.56×10^{1}
Pu-238	5.90×10^{-2}
Ba-137M	3.07
Sr-90	3.12
Ce-144	7.17
Nb-95	4.37
Y-90	3.12
Ru-106	6.11×10^{-1}

F.1.4.2.2.2.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical airplane crash accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence.

Dry Storage Airplane Crash Summary

Site	Probability of accident per year	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
Savannah River	3×10^{-7}	4.7×10^{-1}	2.8	8.4×10^{-7}
Pearl Harbor	1×10^{-5}	19	26	2.6×10^{-4}
Norfolk	1×10^{-6}	72	16	1.6×10^{-5}
Portsmouth	1×10^{-7}	38	9.0	9.0×10^{-7}
Kesselring	1×10^{-7}	7.7	7.5	7.5×10^{-7}
Oak Ridge	3×10^{-7}	120	4.7	1.4×10^{-6}

The risk for this hypothetical accident is most severe at Pearl Harbor and Norfolk. It is also the highest risk for any hypothetical accident evaluated at Pearl Harbor and Norfolk. For the sites with crash probabilities less than 10^{-7} per year, consequences were not calculated since it is expected that they would not substantially contribute to the risk.

For the hypothetical airplane crash into a dry storage cask accident scenario, the radioactive plume might cause contamination of the ground to a downwind distance of approximately 0.9 mile. This would yield a total area impacted by the accident of about 106 acres. The calculated downwind distance would be contained within the boundaries of the Savannah River and Kesselring sites. The contaminated plume would extend beyond the boundaries of Oak Ridge and the shipyards that are at risk for this accident.

Table F.1.4.2.2.2-1. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Savannah River

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.5×10^{1}	5.9×10^{-3}
MCW	8.7×10^{-1}	3.5×10^{-4}
NPA	1.1×10^{-2}	5.5×10^{-6}
MOI	1.8×10^{-1}	8.8×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	9.6×10^{2}	4.8×10^{-1}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.2×10^{1}	7.4×10^{-2}
MCW	1.1×10^{1}	4.4×10^{-3}
NPA	1.9×10^{-1}	9.5×10^{-5}
MOI	4.7×10^{-1}	2.3×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	5.5×10^{3}	2.8

Table F.1.4.2.2.2-2. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Pearl Harbor

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.3×10^{1}	2.7×10^{-2}
MCW	N/A	N/A
NPA	8.6	4.3×10^{-3}
MOI	4.7	2.3×10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	2.0×10^{4}	9.8

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.2×10^{1}	7.4×10^{-2}
MCW	N/A	N/A
NPA	2.8×10^{2}	2.8×10^{-1}
MOI	1.9×10^{1}	9.3×10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 817385	5.2×10^{4}	2.6×10^{1}

Table F.1.4.2.2.2-3. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Norfolk

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.2	3.3×10^{-3}
MCW	N/A	N/A
NPA	2.0	1.0×10^{-3}
MOI	6.9	3.4×10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	6.5×10^{3}	3.2

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.2×10^{1}	7.4×10^{-2}
MCW	N/A	N/A
NPA	2.4×10^{1}	2.4×10^{-2}
MOI	7.2×10^{1}	7.2×10^{-2}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1539002	3.1×10^{4}	1.6×10^{1}

Table F.1.4.2.2.2-4. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Portsmouth

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.1	3.2×10^{-3}
MCW	N/A	N/A
NPA	1.9	9.6×10^{-4}
MOI	3.1	1.6×10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	3.7×10^{3}	1.9

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.2×10^{1}	7.4×10^{-2}
MCW	N/A	N/A
NPA	4.3×10^{1}	4.3×10^{-2}
MOI	3.8×10^{1}	3.8×10^{-2}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 2432627	1.8×10^{4}	9.0

Table F.1.4.2.2.2-5. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Kesselring

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	8.2	3.3×10^{-3}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	1.3	6.6×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	4.8×10^{3}	2.4

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.2×10^{1}	7.4×10^{-2}
MCW	N/A	N/A
NPA	N/A	N/A
MOI	7.7	3.8×10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 1148587	1.5×10^{4}	7.5

Table F.1.4.2.2.2-6. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Oak Ridge

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.3×10^{1}	2.7×10^{-2}
MCW	8.7×10^{-1}	3.5×10^{-4}
NPA	1.1×10^{1}	5.7×10^{-3}
MOI	1.9×10^{1}	9.7×10^{-3}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	2.9×10^{3}	1.4

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.2×10^{1}	7.4×10^{-2}
MCW	5.3	2.2×10^{-3}
NPA	7.2×10^{1}	7.2×10^{-2}
MOI	1.2×10^{2}	1.2×10^{-1}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	9.5×10^{3}	4.7

F.1.4.2.3 Dry Cell Operations.

F.1.4.2.3.1 Inadvertent Cutting into Fuel Region or Mechanical Damage.

F.1.4.2.3.1.1 Description of Conditions. Mechanical damage due to handling during examination, such as accidentally cutting into the fuel region of an element, was assessed. This hypothetical accident results from inadvertent cutting across the fuel region when cropping off the Zircaloy ends of a fuel unit. All noble gas isotopes within the vicinity of the cut might be released to the facility building and escape to the environment. The majority of the volatile and solid nuclides are likely to be retained in the fuel or the facility exhaust filters. The resulting airborne release to the environment was evaluated. The possible exposure to the workers, individuals living on the site boundary, and the general population was evaluated.

F.1.4.2.3.1.2 Source Term. Conditions used in developing the source term are as follows:

- One percent of the fission products in the fuel element being handled are close enough to the cut site to be available for release.
- All (100%) of the noble gases available for release are released to the atmosphere.
- Twenty-five percent of the halogens available for release are released.
- One percent of the particulate fission products could be released and 99.9% of these are removed by normally installed HEPA filters.
- Cs and Ru would behave like particulate fission products.
- The release to the environment occurs at a constant rate over a 15-minute period.
- There is no increase in direct radiation due to this accident.

- The following amounts of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the possible exposure.

Nuclide	Curies
Pu-238	7.2×10^{-5}
Cs-134	2.9×10^{-3}
Cs-137	4×10^{-3}
I-129	2.5×10^{-5}
Sr-90	3.9×10^{-3}
Ce-144	9.0×10^{-3}
Nb-95	5.4×10^{-3}
I-131	5.4×10^{-4}
H-3	1.42
Y-90	3.9×10^{-3}
Ba-137m	3.8×10^{-3}
Ru-106	7.6×10^{-4}
Zr-95	2.9×10^{-3}
Y-91	2.3×10^{-3}
Eu-154	2.7×10^{-4}

F.1.4.2.3.1.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical mechanical damage accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. The probability of damage to fuel during handling is small. The work on fuel at the INEL-ECF includes removal of the non-fueled portions at each end of the fuel unit. This is done in a sawing operation. To cut into the fuel, there must be operator error in positioning the spent fuel in the cutting apparatus and error in selecting the saw cut positioning gage. The combined operator and independent checker error probability for cutting of the fuel has been evaluated to be less than 10^{-7} per cut (Swain and Guttman 1983). Using a conservative number of 10^3 saw cut operations per year results in a fuel cutting probability of less than 10^{-4} per year which has been used in the risk evaluation.

Dry Cell Mechanical Damage Summary			
Site	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
INEL	2.2×10^{-4}	3.5×10^{-4}	3.5×10^{-8}
Savannah River	2.4×10^{-4}	1.4×10^{-3}	1.4×10^{-7}
Hanford	7.1×10^{-5}	5.3×10^{-4}	5.3×10^{-8}
Nevada Test Site	4.0×10^{-4}	3.7×10^{-5}	3.7×10^{-9}
Oak Ridge	5.8×10^{-2}	2.5×10^{-3}	2.5×10^{-7}

The risk for this hypothetical accident is roughly proportional to the surrounding population with Oak Ridge being the worst and the Nevada Test Site being the best.

For the hypothetical dry cell mechanical damage accident scenario, the radioactive plume might result in contamination of the ground to a downwind distance of less than 0.06 mile. This would yield a total area impacted by the accident of less than 0.5 acre. The calculated downwind distance would be contained within the boundaries of all DOE sites under evaluation.

Table F.1.4.2.3.1-1. Summary of Exposure Calculation Results.
For Dry Cell Operations - Mechanical Damage
At INEL

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.7×10^{-2}	1.5×10^{-5}
MCW	3.4×10^{-5}	1.4×10^{-8}
NPA	1.9×10^{-5}	9.5×10^{-9}
MOI	6.2×10^{-5}	3.1×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	3.9×10^{-1}	1.9×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.0×10^{-1}	4.1×10^{-5}
MCW	3.7×10^{-4}	1.5×10^{-7}
NPA	1.1×10^{-4}	5.7×10^{-8}
MOI	2.2×10^{-4}	1.1×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115690	7.0×10^{-1}	3.5×10^{-4}

Table F.1.4.2.3.1-2. Summary of Exposure Calculation Results.
For Dry Cell Operations - Mechanical Damage
At Savannah River

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{-2}	6.6×10^{-6}
MCW	9.6×10^{-4}	3.8×10^{-7}
NPA	1.2×10^{-5}	6.1×10^{-9}
MOI (New ECF)	1.0×10^{-4}	5.1×10^{-8}
MOI (Barnwell)	2.0×10^{-4}	1.0×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	6.2×10^{-1}	3.1×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.0×10^{-1}	4.1×10^{-5}
MCW	1.2×10^{-2}	4.9×10^{-6}
NPA	2.1×10^{-4}	1.0×10^{-7}
MOI (New ECF)	2.4×10^{-4}	1.2×10^{-7}
MOI (Barnwell)	1.7×10^{-3}	8.4×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	2.8	1.4×10^{-3}

Table F.1.4.2.3.1-3. Summary of Exposure Calculation Results.
For Dry Cell Operations - Mechanical Damage
At Hanford

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{-2}	6.6×10^{-6}
MCW	1.3×10^{-3}	5.1×10^{-7}
NPA	1.5×10^{-5}	7.4×10^{-9}
MOI (New ECF)	9.8×10^{-6}	4.9×10^{-9}
MOI (FMEF)	2.0×10^{-5}	9.9×10^{-9}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	6.2×10^{-2}	3.1×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.0×10^{-1}	4.1×10^{-5}
MCW	8.0×10^{-3}	3.2×10^{-6}
NPA	2.3×10^{-4}	1.2×10^{-7}
MOI (New ECF)	7.1×10^{-5}	3.6×10^{-8}
MOI (FMEF)	2.5×10^{-4}	1.2×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375860	1.07	5.3×10^{-4}

Table F.1.4.2.3.1-4. Summary of Exposure Calculation Results.
For Dry Cell Operations - Mechanical Damage
At Nevada Test Site

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.8×10^{-3}	2.3×10^{-6}
MCW	4.5×10^{-6}	1.8×10^{-9}
NPA	N/A	N/A
MOI	4.7×10^{-5}	2.3×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	3.6×10^{-2}	1.8×10^{-5}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.0×10^{-1}	4.1×10^{-5}
MCW	2.6×10^{-4}	1.0×10^{-7}
NPA	N/A	N/A
MOI	4.0×10^{-4}	2.0×10^{-7}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	7.4×10^{-2}	3.7×10^{-5}

Table F.1.4.2.3.1-5. Summary of Exposure Calculation Results.
For Dry Cell Operations - Mechanical Damage
At Oak Ridge

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	3.7×10^{-2}	1.5×10^{-5}
MCW	9.6×10^{-4}	3.8×10^{-7}
NPA	1.3×10^{-2}	6.3×10^{-6}
MOI	9.3×10^{-3}	4.6×10^{-6}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	1.9	9.5×10^{-4}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.0×10^{-1}	4.1×10^{-5}
MCW	5.9×10^{-3}	2.4×10^{-6}
NPA	8.0×10^{-2}	4.0×10^{-5}
MOI	5.8×10^{-2}	2.9×10^{-5}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	5.1	2.5×10^{-3}

F.1.4.2.3.2 Partial Loss of Shielding Due to Earthquake.

F.1.4.2.3.2.1 Description of Conditions. A hypothetical earthquake causes the proposed Dry Cell Facility to lose some portion of its concrete shielding. Direct radiation exposure to the on-site work force and the general public has been calculated.

F.1.4.2.3.2.2 Source Term. The conditions used to calculate the dry cell direct radiation levels are as follows:

- For calculational purposes, a total of 50% of the high-density concrete dry cell shielding might be removed due to the earthquake. More realistic damage from an earthquake would result in cracks or small openings in the shielding. This bounds anticipated damage to the facility.

- Building containment and ventilation systems remain in operation. Therefore, there is no airborne release to the environment. Calculations have already been performed in Section F.1.4.2.1.1 for a drained water pool hypothetical accident which bound any anticipated airborne releases from the dry cell facility should the building containment and ventilation systems fail.

F.1.4.2.3.2.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical loss of shielding accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. As discussed in Section F.1.4.2.1.1.3, the probability of this hypothetical accident is estimated to be 10^{-5} per year.

Dry Cell Partial Loss of Shielding Summary			
Site	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
INEL	9.3×10^{-17}	3.0×10^{-19}	3.0×10^{-24}
Savannah River	6.7×10^{-15}	3.0×10^{-16}	3.0×10^{-21}
Hanford	3.3×10^{-23}	4.9×10^{-24}	4.9×10^{-29}
Nevada Test Site	6.3×10^{-11}	3.7×10^{-37}	3.7×10^{-42}
Oak Ridge	1.2×10^{-2}	7.5×10^{-6}	7.5×10^{-11}

At all sites, the risks associated with this accident are the lowest of any accident evaluated.

Table F.1.4.2.3.2-1. Summary of Exposure Calculation Results. For Dry Cell Operations - Partial Loss of Shielding At INEL

Receptor Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.2×10^{-5}	2.9×10^{-8}
MCW	7.5×10^{-13}	3.0×10^{-16}
MOI	9.3×10^{-17}	4.7×10^{-20}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 115,690	5.9×10^{-16}	3.0×10^{-19}

Table F.1.4.2.3.2-2. Summary of Exposure Calculation Results. For Dry Cell Operations - Partial Loss of Shielding At Savannah River

Receptor Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.2×10^{-5}	2.9×10^{-8}
MCW	2.7×10^{-6}	1.1×10^{-9}
MOI (New ECF)	6.7×10^{-15}	3.4×10^{-18}
MOI (Barnwell Plant)	2.4×10^{-6}	1.2×10^{-9}
NPA	7.9×10^{-17}	4.0×10^{-20}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579,541	5.9×10^{-13}	3.0×10^{-16}

Table F.1.4.2.3.2-3. Summary of Exposure Calculation Results.
For Dry Cell Operations - Partial Loss of Shielding
At Hanford

Receptor Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.2×10^{-5}	2.9×10^{-8}
MCW	2.7×10^{-6}	1.1×10^{-9}
MOI (New ECF)	3.3×10^{-23}	1.7×10^{-26}
MOI (FMEF)	6.7×10^{-15}	3.4×10^{-18}
NPA	3.9×10^{-25}	2.0×10^{-28}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 375,860	9.7×10^{-21}	4.9×10^{-24}

Table F.1.4.2.3.2-4. Summary of Exposure Calculation Results.
For Dry Cell Operations - Partial Loss of Shielding
At Nevada Test Site

Receptor Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.2×10^{-5}	2.9×10^{-8}
MCW	7.1×10^{-15}	2.8×10^{-18}
MOI	6.3×10^{-11}	3.2×10^{-14}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 12,159	8.7×10^{-33}	4.4×10^{-36}

Table F.1.4.2.3.2-5. Summary of Exposure Calculation Results.
For Dry Cell Operations - Partial Loss of Shielding
At Oak Ridge

Receptor Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	7.2×10^{-5}	2.9×10^{-8}
MCW	5.5×10^{-7}	2.2×10^{-10}
MOI	1.2×10^{-2}	6.0×10^{-6}
NPA	1.4×10^{-4}	7.0×10^{-8}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871,531	1.5×10^{-2}	7.5×10^{-6}

F.1.4.2.3.3 Airplane Crash Into Dry Cell Facility.

F.1.4.2.3.3.1 Description of Conditions. A hypothetical aircraft accident scenario was developed for dry cell operations. Based on the probability of occurrence, as discussed in Section F.3, specific analysis was only performed for Savannah River, the Nevada Test Site, and Oak Ridge. The accident was postulated to cause major damage to the building, resulting in the loss of containment and filtered exhaust systems. The fuel units inside the dry cell could also be damaged due to mechanical impacts and potential fire. The fission products which might be released are based on factors derived from overheating testing performed on the naval fuel systems. The mechanical impact also could result in the release of corrosion products to the environment.

F.1.4.2.3.3.2 Source Term. The development of the radioactive source term for this scenario is based on the following:

- One percent of the fuel units stored inside of the dry cell might be damaged by either the impact or resultant fire and those fission products would be available for release.

- Of the fission products available for release, 100% of the noble gases, 3% of the halogens, 1.1% of the cesium, and 0.1% of the remaining solids could be released to the environment.

- The release to the environment would occur at a constant rate over a 15-minute period.

- 10% of the available corrosion products could be released to the environment.

- A portion of the concrete shielding is destroyed; however, the resultant rubble provides a minimum of 6 inches of concrete shielding.

- The following amount of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the possible exposure.

Nuclide	Curies
Cs-134	4.5×10^{1}
Cs-137	6.23×10^{1}
Pu-238	1.03×10^{-1}
BA-137M	5.37
Sr-90	5.46
Ce-144	1.25×10^{1}
Nb-95	7.65
Y-90	5.46
Ru-106	1.07

F.1.4.2.3.3.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical airplane crash into the dry cell at the Savannah River Site. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence.

Site	Probability of accident per year	Maximally exposed off-site individual (MOI) (rem)	No. of fatal cancers if accident occurs	Risk per year
Savannah River	2×10^{-6}	8.2×10^{-1}	4.8	9.6×10^{-6}
Nevada Test Site	4×10^{-7}	1.6	1.8×10^{-1}	7.2×10^{-8}
Oak Ridge	1×10^{-6}	350	8.4	8.4×10^{-6}

This accident results in the highest risk for any hypothetical accident evaluated at Savannah River, the Nevada Test Site, and Oak Ridge.

For the hypothetical airplane crash into a dry cell accident scenario, the radioactive plume might cause contamination of the ground to a downwind distance of approximately 1.3 miles. This would yield a total area impacted by the accident of about 207 acres. The calculated downwind distance would be contained within the boundaries of Savannah River and the Nevada Test Site, but not Oak Ridge.

Table F.1.4.2.3.3-1. Summary of Exposure Calculation Results.
For Dry Cell Operations - Airplane Crash
At Savannah River

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	2.6×10^{1}	2.1×10^{-2}
MCW	1.6	6.2×10^{-4}
NPA	1.9×10^{-2}	9.6×10^{-6}
MOI	3.1×10^{-1}	1.5×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	1.6×10^{3}	8.1×10^{-1}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{2}	1.3×10^{-1}
MCW	1.9×10^{1}	7.8×10^{-3}
NPA	3.3×10^{-1}	1.7×10^{-4}
MOI	8.2×10^{-1}	4.1×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 579541	9.6×10^{3}	4.8

Table F.1.4.2.3.3-2. Summary of Exposure Calculation Results.
For Dry Cell Operations - Airplane Crash
At Nevada Test Site

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	9.2	3.7×10^{-3}
MCW	7.1×10^{-3}	2.9×10^{-6}
NPA	N/A	N/A
MOI	2.5×10^{-1}	1.3×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	2.1×10^{2}	1.1×10^{-1}

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{2}	1.3×10^{-1}
MCW	4.2×10^{-1}	1.7×10^{-4}
NPA	N/A	N/A
MOI	1.6	8.0×10^{-4}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 13792	3.5×10^{2}	1.8×10^{-1}

Table F.1.4.2.3.3-3. Summary of Exposure Calculation Results.
For Dry Cell Operations - Airplane Crash
At Oak Ridge

50% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	5.8×10^{1}	4.7×10^{-2}
MCW	1.5	6.2×10^{-4}
NPA	2.2×10^{1}	2.2×10^{-2}
MOI	1.7×10^{2}	1.7×10^{-1}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	5.2×10^{3}	2.6

95% METEOROLOGY		
Location	Total EDE (rem)	Likelihood of Fatal Cancer
Worker	1.6×10^{2}	1.3×10^{-1}
MCW	9.3	4.7×10^{-3}
NPA	1.3×10^{2}	1.3×10^{-1}
MOI	3.5×10^{2}	3.5×10^{-1}
Exposure to Population within 50-mile Radius (person-rem)		Number of Fatal Cancers
Population of 871531	1.7×10^{4}	8.4

F.1.4.3 Impact of Accidents on Close-in Workers. An evaluation has been made of the impact to close-in workers involved in naval spent nuclear fuel management that might occur due to the various radiological accidents postulated in spent fuel handling. This evaluation focused on the radiological consequences of the accident. Clearly, a limited number of fatalities may occur which are related to spent fuel handling only in a secondary manner; i.e., the worker who happened to be in the facility may be killed due to a plane crash, seismic event, crane failure, etc. These secondary effects are not discussed in the following. Rather, only radiological consequences are considered.

F.1.4.3.1 Wet Storage.

F.1.4.3.1.1 Drained Water Pool Due to Seismic Event. No fatalities to workers close to the scene of the accident would be expected due to radiological consequences. This is because drainage of the large amount of water in a water pool is expected to take several days which provides ample time for workers to leave the facility.

F.1.4.3.1.2 Accidental Criticality in a Water Pool Due to Human Error. It is likely no fatalities would occur. At most, two or three workers may receive some appreciable radiation exposure. This is because the criticality would occur under approximately 20 feet of water. Shielding by the water would be sufficient to prevent exposure of nearby workers. Expulsion of a cone of water above the criticality might lead to significant exposure to any workers who were directly above the location of the criticality.

F.1.4.3.1.3 Mechanical Damage to Fuel in a Water Pool Due to Operator Error or Crane Failure. No fatalities to workers would be expected from radiological consequences. This is because the release of the source term is underwater. Attenuation by the water would occur for most products, but release of noble gases would cause a direct radiation exposure to workers in the area. Upon releases from the surface of the water pool, radiation alarms would sound requiring evacuation of nearby workers. Timely evacuation would prevent substantial radiation exposure.

F.1.4.3.1.4 Airplane Crash into Water Pool Storage. No fatalities to workers would be expected from radiological consequences. This is because any release of radioactive products would be underwater and radiation alarms would sound requiring evacuation of nearby workers. Timely evacuation would prevent substantial radiation exposure.

F.1.4.3.2 Dry Storage.

F.1.4.3.2.1 Wind-driven Missile Impact on Storage Casks. It is likely there would be no fatalities to workers from radiological consequences. This is because there usually would be no nearby workers except for brief periods when a container is being placed in the dry storage array. Since a wind-driven missile is not expected to penetrate a dry storage container, direct radiation exposures even to nearby workers would not be expected. The container seal could be breached and some airborne products released. At most, two or three nearby workers may receive some radiation exposure from inhalation of airborne radioactivity.

F.1.4.3.2.2 Airplane Crash into Dry Storage. It is not likely that any fatalities would occur to nearby workers due to the radiological consequences of this accident. As in Section F.1.4.3.2.1 above, workers are usually not in the dry storage array except when a container is being placed into the array. At most, two or three nearby workers might receive significant radiation exposure from inhalation of airborne radioactivity since the container seal may be breached. The low probability of the airplane crash itself, coupled with the probability that workers would be close enough to be affected, coupled with the probability that the wind would be blowing in the direction of the workers, makes it very unlikely that any worker would receive substantial radiation exposure.

F.1.4.3.3 Dry Cell Operations.

F.1.4.3.3.1 Inadvertent Cutting into Fuel or Mechanical Damage. No fatalities to workers would be expected from the radiological consequences of this accident. This is because the ventilation systems' exhaust from a dry cell is directed to the outside of the building in which a dry cell is constructed and away from nearby workers.

F.1.4.3.3.2 Partial Loss of Shielding of a Dry Cell. It is likely that no fatalities would occur among nearby workers from the radiological consequences of this accident. This is because there is still substantial shielding of radiation from material inside the cell even with the assumed 50-percent loss of the high-density concrete. However, one or two nearby workers may receive some exposure from radiation streaming through a crack in the dry cell if this is the mode of failure. Workers are trained to evacuate quickly when radiation alarms sound.

F.1.4.3.4 Other Accidents.

F.1.4.3.4.1 HEPA Filter Fire. No fatalities would be expected among nearby workers from the radiological consequences of a fire in a HEPA filter. This is because HEPA filters are not located in an area where workers are likely to be working. In addition, the release of radioactivity involved in a HEPA filter fire is not large.

F.1.4.3.4.2 Small Leaks from Water Pools. No fatalities are expected among nearby workers from the radiological consequences of a small leak from a water pool. The leak would be expected to be into the ground through the water pathway. Drinking water supplies would not be immediately impacted. In addition, the typical concentration of radioactivity in the water is low.

F.1.4.4 Evaluation of Shipboard Fire Involving Shipping Containers.

F.1.4.4.1 Description of Conditions. In this hypothetical accident scenario, a fire onboard a ship that is transporting naval spent nuclear fuel in shipping containers from Pearl Harbor to Puget Sound is postulated. This accident could be initiated by a collision with another ship. The collision and subsequent fire are postulated to occur in Puget Sound in the center of the shipping lane at a distance of approximately 2 miles from Seattle. The consequences of a similar accident at Pearl Harbor would be less because of the smaller population and the fact that Pearl Harbor is a restricted area and is very close to the sea on the south side, limiting the number of people who might be exposed. This section addresses the radiological consequences of this postulated accident scenario. The toxic chemical consequences related to the burning fuel oil are presented in Section F.2.4.2.2.

During shipment, the containers are well protected from direct mechanical damage should a ship collision occur. The rugged nature of the shipping container and the naval reactor's fuel system is demonstrated by the analysis of airplane crashes which showed that a jet engine rotor would not penetrate the container or rupture the fuel. A severe fire is necessary to potentially cause failure of the container seals and overheat the spent fuel sufficiently to release fission products. Collisions of this severity are extremely unlikely. During the hypothetical accident, the fire would need to burn intensely in the hold for several hours to cause release of fission products or corrosion products to the environment.

F.1.4.4.2 Source Term. Conditions used in developing the source term are as follows:

- Ten percent of all fuel unit cladding inside of two shipping containers is ruptured and the contained fission products are available to be released from the fuel units.
- Of the available fission products, 100% of the noble gases, 3% of the halogens, 1.1% of the cesium, and 0.1% of the remaining solid fission products are assumed to be released to the container.
- Ten percent of all fission products released to the container are released to the environment and the remainder are adherent on the fuel and cask surfaces.
- Ten percent of the original corrosion products from the fuel units are released from the cask to the environment.
- The following amount of radionuclides could be released to the environment. This listing includes nuclides from one container that result in at least 99% of the possible exposure.

Nuclide	Curies
Cs-134	2.57×10^{1}
Cs-137	3.56×10^{1}
Pu-238	5.90×10^{-2}
Ba-137M	3.07
Sr-90	3.12
Ce-144	7.17
Nb-95	4.37
Y-90	3.12
Ru-106	6.11×10^{-1}

F.1.4.4.3 Results. The following table summarizes the public health risk to the general population that would result from the hypothetical shipboard fire accident. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence.

The probability of occurrence of this hypothetical shipping accident is 6.7×10^{-8} per year or less, and was obtained as follows. The probability of a single port entry accident is 1.6×10^{-4} (DOE 1994). The probability of a fire, given the occurrence of an accident, is 8×10^{-4} (DOE 1994). Combining these two probabilities with the port entry frequency of 21 naval spent nuclear fuel shipments spread over 40 years results in a probability of 6.7×10^{-8} per year. Due to the rugged nature of the naval fuel and likely effectiveness of fire fighting over a several hour period, the probability of fission product release to the environment would be even less.

DOE guidance (DOE 1993b) provides that the consequence of an accident which has a probability of occurrence of less than 1×10^{-7} per year need not be calculated. However, in view of interest in this accident expressed in several public comments, the following table is provided listing both the consequence and the risk.

Shipboard Fire Involving Shipping Containers				
In Puget Sound Shipping Lane	50% Meteorology		95% Meteorology	
Maximally Exposed Off-site Individual (MOI)	Total EDE (Rem)	Likelihood of Fatal Cancer	Total EDE (Rem)	Likelihood of Fatal Cancer
	9.3×10^{-1}	4.7×10^{-4}	1.8	9.2×10^{-4}
General Population within 50-mile Radius	Exposure (Person-Rem)	Number of Fatal Cancers	Exposure (Person-Rem)	Number of Fatal Cancers
	2.27×10^{4}	11.4	1.03×10^{5}	51.5
Risk per year	7.6×10^{-7}		3.5×10^{-6}	

The risk for this hypothetical accident is slightly lower than that for the most severe facility accident analyzed at Puget Sound.

For the hypothetical shipboard fire accident, the radioactive plume might cause contamination to a downwind distance of less than 1 mile. However, since this area is entirely over water, the contamination would be quickly diluted by tidal flow and turbulence.

F.1.5 Analysis of Uncertainties

The analyses of the impacts of normal operations and hypothetical accidents associated with management of naval spent nuclear fuel presented in this Environmental Impact Statement (EIS) are based on conservative calculations. This is necessary because virtually all of the events analyzed have never occurred and most of the impacts of routine operations are so small that they cannot be measured. The use of calculations introduces the possibility that the actual impacts may differ from those calculated due to various kinds of uncertainties, such as differences between actual behavior and the theoretical models or equations and the variability of the values of factors used in the calculations. In order to portray the effects of such variability and uncertainty, the analyses performed for this appendix have been divided into four components: the probability that an event, such as an accident, could occur; the amount of radioactive material or radiation that might be released by the event; the calculation of the potential for exposure to human beings from the release; and the conversion of the radiation exposure to detrimental health effects. Each of these components is discussed separately in the following sections for both routine operations and accidents.

Each of these components has been analyzed for both routine operations and accidents. The discussion in the following sections focuses on accident analyses, but it should be understood that the analysis of uncertainties for routine operations is the same, with a few exceptions. First, routine operations are certain to occur, so the "probability" of such events is effectively 1.0. Second, the source terms used for the analyses of routine operations are based on monitoring of current operations at Naval Nuclear Propulsion Program facilities such as the Expended Core Facility at INEL. Consequently, the estimates of the amount of radiation or radioactivity involved are expected to be close to those which might actually occur under the alternatives evaluated in this EIS. It is possible that there would be some variations among facilities and that future efforts to keep exposures to workers as low as reasonably achievable might reduce the source terms further, but the values used in the analyses in this EIS are expected to be little different from those actually encountered. The effects of routine operations and accidents have been calculated using similar analytical methods and models for determination of radionuclide movement in the environment, pathways to humans, and conversion of exposure to health effects. Therefore, the discussion of uncertainties in Sections F.1.5.3 and F.1.5.4 applies to the results of analyses of routine operations, as well as to postulated accidents.

F.1.5.1 Probabilities of Events. The probability that an accident might occur has been determined for a number of events which might reasonably be postulated. These probabilities are used in this appendix to calculate the risk, defined as the product of the probability times the consequences, for each postulated accident.

The best methods available have been used to estimate the probabilities for the events selected for analysis. For example, a methodology developed by Sandia Laboratories (Sandia 1983) was used to compute the probability that an aircraft might crash into naval spent nuclear fuel facilities. This method uses actual aircraft crash statistics obtained from the Federal Aviation Administration and was developed by Sandia to reproduce the observed frequencies as closely as possible. Probabilities for seismic events were derived from published studies of the frequencies of seismic activity and represent the best available estimates, but these probabilities are subject to some uncertainty due to the relatively few events which have occurred at the sites evaluated under the alternatives in this EIS.

The probabilities of a range of accidents which might be caused by human error have also been included. Such events include accidental criticality caused by handling errors, dropping of fuel modules, improper operation of cranes, and incorrectly performing machining procedures. For human error, a probability of one error in one thousand operations (a frequency of 10^{-3} events per year) is used for operations performed by a single trained operator following a written procedure. If the procedure requires verification of the action by a second trained operator, this frequency is lowered to 10^{-4}. These probabilities are derived from the methodology used by the Nuclear Regulatory Commission for assessment of human reliability (Swain 1983).

In many instances, the probabilities assigned to the events reflect the likelihood that a particular event, such as an earthquake or an aircraft crash, might occur. However, for the purpose of the analyses, the resulting accident was assumed to have quite severe consequences. The probability of such severe consequences is smaller than the probability that the initiating event might occur, with consequences as severe as used in the analyses possibly occurring only one time in 10 or 100 occurrences of the initiating event. The probabilities for most of the analyses in this appendix used only the probability of the initiating event and did not include the further reduction in the probability of the postulated severe consequences resulting from the severity used. This was done, in part, because the severe consequences assumed, and in some cases the initiating events themselves, occur very infrequently, or have never occurred, so little data on their frequency is available.

For example, one accident analyzed is the impact on a spent fuel container of a missile produced by a tornado or other high winds. The sequence of events analyzed included breaching the container seal in order to release radioactive material. In reality, the missile would have to be large enough and traveling at high enough speed to cause the postulated damage. Similarly, it would have to contact the container at the correct location and at the correct angle in order to damage the seal. The probability assigned to this accident is 10^{-5} per year, the probability that a wind-driven missile might strike a container, and does not include any factor to account for other elements in the sequence required to actually damage the seal. Therefore, the probability of the consequences calculated for this accident would be much smaller than the probability of 10^{-5} per year used in the analysis.

A second example is provided by the analysis of aircraft impact on shipping containers used for storage of naval spent nuclear fuel. In this accident analysis, the impact was assumed to cause a shipping container to be penetrated if the container were contacted by the aircraft. However, naval spent nuclear fuel shipping containers are of very rugged design, and structural analysis of the container showed that a naval shipping container is very unlikely to be penetrated by an aircraft crash, even by the hardest parts of the airplane. Consequently, the probability that the naval spent nuclear fuel could be damaged and that fission products might be released is much, much less than the crash probability alone, which is the probability assigned to these consequences in this appendix.

A third example is seen in the ship fire accident. In this analysis, it is assumed that if a ship carrying naval spent nuclear fuel shipping containers were involved in a very severe collision and a fire occurred, the fire would include the cargo hold where the naval spent nuclear fuel containers are carried, the fire would not be extinguished by the redundant systems provided, and it would burn long enough at sufficient intensity to damage the shipping container and the spent nuclear fuel inside and cause release of radioactive materials from the containment provided. Given that a severe collision occurred, the probability that all of the necessary conditions would occur and a fire of the required intensity and duration would occur in the cargo hold is clearly far less than the probability of the collision.

As can be see from these examples, the actual probability of the consequences resulting from the analyses are smaller than the values presented in this appendix, at least in part because these probabilities do not include an additional factor to reflect the accident severity used in the analyses. As a result, the risks stated in this appendix for most accidents are believed to be at least 10 to 100 times larger than what would actually occur. However, the same probabilities have been used in the

evaluation of all of the alternatives considered and all of the risks are small, so the approach used is adequate for the purposes of this EIS.

F.1.5.2 Release of Radioactive Material or Radiation (Source Term). Since the source terms used in the accident analyses are typically for accidents which have never occurred, there is greater room for uncertainty. All of the accidents analyzed in this EIS are intended to be accidents which produce consequences which are unlikely to be exceeded by any reasonably foreseeable accident. As a result, the accidents themselves and the sequences of events during the accidents have been chosen to maximize the source term. For example, systems such as high efficiency particulate filters have been considered to be inoperative in all cases where the accident might have an opportunity to disable them.

The source terms for the hypothetical accident analyses are dependent upon a number of factors. For there to be an accidental release of radioactivity to the environment, there must be damage to the storage facility or containment structure. Furthermore, naval spent nuclear fuel must be damaged as well in order for there to be any release of fission products since all fission products are fully contained within naval nuclear fuel. The amount of damage to the external containment or the fuel is dependent upon the severity and the nature of the accident. In the accidents analyzed, there are assumptions concerning the containment or the extent of damage to the fuel units which were made to provide a conservative, bounding evaluation whose results would not be exceeded by reasonably postulated accidents of a similar type.

One example of this is the evaluation of the dry storage container impacted by a wind-driven missile. Damage to the container by the missile is not expected to occur, but for the analysis in this EIS, the seal is assumed to be damaged by the missile impact and corrosion products within the container are assumed to be released through the damaged seal. The uncertainty on the resultant release is one-sided since the probability of a release larger than in the calculation (resulting in a higher calculated dose) is essentially zero while the possibility of a release of less radioactive material is large (for example, no release if the container seal is not broken). The range of variation, or the uncertainty interval, in the source term for this accident is between +0% and -100%.

Another example is the plane crash into a dry processing facility for naval spent nuclear fuel. The dry processing facility includes a thick concrete shielded cell in which a few naval spent nuclear fuel units are processed at a time. The massive concrete shield is provided to protect operating personnel from radiation but it has the secondary benefit of protecting the fuel units being processed from missiles caused by natural or man-made phenomena. In the unlikely event that an airplane crashed into the facility, it is expected that no damage to the spent fuel would result. Even so, for evaluation of this accident in this EIS, it is assumed that 1% of the fuel in the dry cell could be damaged and that sufficient jet fuel could enter the dry cell to cause a fire which could cause the release of fission products from the damaged fuel and destroy the filtration system. Again, the uncertainty range is one-sided since no damage to fuel is expected, causing the variability or uncertainty to range from +0% to -100%.

All of the source terms used for the evaluation of the accidents were developed in a similar fashion. Thus, the expected outcome for all of the accidents is that a lower release to the environment is expected than is used in the analysis, representing a range of variation of +0% to -100%.

F.1.5.3 Exposure to Humans. Exposure to the individuals and the general population is evaluated by integrated computer programs. The methods used model the movement of airborne, ground, and water contamination resulting from the postulated release using five types of pathways to the population. These pathways include exposure directly to the radiation from the material in the plume, direct exposure to radiation from contaminated soil or water, inhalation of air containing gases or particles, and ingestion of contaminated water or food. The analyses in this appendix used parameter values which were the best available estimates or, when best estimate values were not available, are conservative.

The Gaussian plume model used in these analyses to represent airborne movement of radioactive material is the standard used in virtually all evaluations of environmental effects. Comparison of distributions calculated using the Gaussian plume model with test data has shown that the results may differ by as much as a factor of 5 in some circumstances. In order to ensure that exposures would be as high as could occur under any set of conditions, in most of the analyses a ground level release was used and no reduction in the airborne concentrations was included for either turbulence caused by buildings or the effect of wind meander which occurs naturally at the low wind speeds accompanying the worst case meteorological conditions.

One intentional choice of parameters to ensure that the results would be conservative is the use of the worst case meteorological conditions in the tabulations of the risks and consequences for all alternatives provided in Chapters 3 and 5. The results for both the most likely meteorological

conditions and for the worst case are provided in detailed tables in this attachment and show that the worst case meteorological conditions produce exposure estimates which are 2 to 10 times higher than those for the most likely conditions (depending upon local meteorological conditions). Overall, the net effect is that the Gaussian plume model might introduce an uncertainty of a factor of 5 or less in either direction, but the use of the worst case meteorological conditions would essentially offset any underestimation of effects.

The direct radiation from the cloud is calculated using a conservative representation of the plume as a finite cloud, and, as a result, little uncertainty is introduced in this part of the analysis. Direct radiation from contamination which results from particles from the plume deposited on the ground surface depends upon the deposition parameters which are input as best-estimate values. Faster deposition would result in more material on the ground and increased exposure to those closer to the accident location but less material on the ground and decreased exposure for those farther from the accident site. Any effects of uncertainty in this parameter would depend upon the population distribution around the postulated accident scene.

The possible exposure to direct radiation from material in surface water and associated sediments as a result of accidental release directly to the water or fallout from an airborne release was estimated for people involved in activities such as professional fishing, maritime operations, swimming, and boating. The calculations took no credit for dilution by river currents or tidal movement and the concentrations in the air were not reduced by the amount of material deposited in the water. Due to the conservative concentrations used in the calculations and an assumption that every member of the population in the area would be exposed to direct radiation from surface waters, exposure from this pathway is very likely overestimated.

The inhalation pathway evaluation is based on average breathing rates and uptake consistent with the recommendations by the ICRP (ICRP 1977 and ICRP 1979). Obviously, higher values for these parameters would increase the estimated exposures and lower values would decrease the estimates. There appears to be little controversy concerning these parameters and the same parameters are used for evaluation of all of the alternatives in this appendix.

The ingestion pathway includes meat, seafood, dairy and crop products, and drinking water. Best-estimate parameters are used to evaluate the contamination levels in food and water when ready for consumption. Consumption rates for individuals are based on observed eating habits. The analysis also includes the assumption that a conservative 10% of the entire diet of the affected population consists of contaminated products. The uncertainties associated with these pathways can obviously affect the estimated impacts, but the range of variation is not large and the same values for a given site were used for evaluation of all alternatives.

The drinking water contribution to the ingestion pathway was calculated by assuming that a portion of the radioactive material would become dissolved in the drinking water supply. At sites where fresh surface water provides drinking water, any contamination of the water was assumed to occur promptly and no decreases due to radioactive decay were used. At sites where aquifers are a source of drinking water, consumption of water from the aquifer was delayed for the time required for the contamination to reach the aquifer and then to reach the nearest drinking water source. As an example, for a postulated leak from the Expended Core Facility, it was assumed that 20 years would pass before water carrying the radioactive material would reach a well drawing from the aquifer and that 1 percent of material released would enter the aquifer each year. Maximum exposed individuals were conservatively assumed to drink only water from the contaminated source and to drink 2 liters of water per day. For the population in general, a conservative fraction of the population was assumed to drink 1 liter of water per day from affected sources. The concentrations in these calculations are considered to be higher than expected because no reduction of the concentration by dilution was included and the fraction of the population exposed to the affected drinking water is conservatively high.

At sites where irrigation is used, contamination of food crops, livestock, and local game was analyzed. The same concentration of radioactive material as in drinking water was used in the irrigation water. Affected crops, livestock, and game were assumed to receive all water from the contaminated water source and applicable biological accumulation factors were used. Human consumption rates for the crops, livestock, and game were used to calculate the exposure from this source. The uncertainty from this source is associated with the concentration of contaminants in the irrigation water, the amount of such foods consumed, and the fraction of the population which ingests the affected food.

The population used to determine the effects of postulated accidents in this appendix is the entire population within the 22.5-degree sector at each distance within 50 miles downwind of the accident. The spread of the plume for the worst case meteorology does not cover the entire sector. The result is that there is a conservatism of more than a factor of 2 in the application of the

calculations to the evaluation of the dose to the population. The population data used were obtained from the 1990 U. S. census, so population growth or decreases in a region could introduce small changes, but the same population distributions were used for a specific site for evaluation of all alternatives.

Considering all of the factors which might have an appreciable effect on the results of the analyses, any tendency of the Gaussian plume model to underestimate concentrations would be offset by the use of other parameters which are known to be conservative. Examples of such conservative factors include the general use of the meteorological conditions which would produce the most severe effects and the use of the entire population of a 22.5-degree sector. Consequently, this portion of the analyses would appear to contribute little in the way of uncertainty which could cause the results to be greater than presented in this appendix.

F.1.5.4 Conversion of Exposure to Health Effects. The conversion of amounts of radiation or radioactive material transmitted to an individual or to population groups requires the calculation of the exposure or dose received by humans caused by inhaling or ingesting radioactive material or by being in a radiation field. Such calculations are based on a number of factors, including the nature and rate of human metabolic processes, such as respiration or excretion, the type of radiation involved, the sensitivity of various organs, and the age of the individuals involved. The rates of human metabolic processes are well characterized at this time and the energies, half-lives, and similar properties of radioactive material or radiation have been measured extensively and are not subject to great debate. Consequently, these factors introduce little uncertainty into the calculations in this EIS.

However, the number of detrimental health effects which might result from exposure of a large group of people to low levels of radiation has been the subject of debate for many years. The National Academy of Sciences has conducted several investigations of this matter and its full commentary on page 181 of its latest study of the health effects of exposure to low levels of radiation, frequently identified as BEIR V (NAS 1990), states:

> Finally, it must be recognized that derivation of risk estimates for low doses and dose rates through the use of any type of risk model involves assumptions that remain to be validated. At low doses, a model dependent interpolation is involved between the spontaneous incidence and the incidence at the lowest doses for which data are available. Since the committee's preferred risk models are a linear function of dose, little uncertainty should be introduced on

this account, but departure from linearity cannot be excluded at low doses below the range of observation. Such departures could be in the direction of either an increased or decreased risk. Moreover, epidemiologic data cannot rigorously exclude the existence of a threshold in the millisievert dose range. Thus, the possibility that there may be no risks from exposures comparable to external natural background radiation cannot be ruled out. At such low dose rates, it must be acknowledged that the lower limit of the range of uncertainty in the risk estimates extends to zero.

The National Academy of Sciences considers that the uncertainty in the lifetime total excess cancer mortality risk estimates calculated using the linear extrapolation, no threshold models it has designated as preferred, which is consistent with the model used in this EIS, is approximately a factor of 2 in either direction (an interval of 0.5 to 2 times the calculated estimates).

The calculations of health effects performed in this Environmental Impact Statement use the relation recommended by the International Council on Radiation Protection because it is well-documented and kept up to date by the Council. It is also consistent with the preferred model identified by the National Academy of Sciences in the BEIR V report and is widely accepted by the scientific community as representing a method which produces estimates of health effects which will not be exceeded. However, there are some who believe that exposure to low levels of radiation can produce more health effects than would be estimated using the International Council on Radiation Protection relation. On the other hand, a growing number of researchers believe that the International Council on Radiation Protection relation overestimates the number of detrimental health effects produced by low levels of radiation and, in fact, the possibility of no effect cannot be excluded (CIRRPC 1992).

Clearly, using a relation developed by one or the other of these groups would produce a larger or smaller estimate of the number of health effects than the values presented in this EIS, but a factor of 2 change in the small risks calculated for all of the alternatives would still leave them as small risks. All of the results of analyses of normal operations and hypothetical accidents in Appendix D include the calculated exposure in addition to the number of health effects in order to permit independent calculations using any relation between radiation exposure and health effects judged appropriate.

F.1.5.5 Summary of Uncertainties. As discussed in the preceding portions of this section, the calculations in this EIS have generally been performed in such a way that the estimates of risk provided are unlikely to be exceeded during either normal operations or in the event of an accident. For routine operations, the results of monitoring of actual operations provide clearly realistic source terms, which, when combined with conservative estimates of the effects of radiation, produce estimates of risk which are very unlikely to be exceeded. The effects for all alternatives have been calculated using the same source terms and other factors, so this EIS provides an appropriate means of comparing potential impacts on human health and the environment.

The analyses of hypothetical accidents provide more opportunities for uncertainty, primarily because the calculations must be based on sequences of events and models of effects which have not occurred. In this appendix, the goal in selecting the hypothetical accidents analyzed has been to evaluate events which would produce effects which would be as severe or more severe than any other accidents which might reasonably be postulated. The models have attempted to provide estimates of the probabilities, source terms, pathways for dispersion and exposure, and the effects on human health and the environment which are as realistic as possible. However, in many cases, the very low probability of the accidents postulated has required the use of models or values for input which produce estimates of consequences and risks which are higher than would actually occur because of the desire to provide results which will not be exceeded. In summary, it is judged that the risks presented in this appendix are believed to be at least 10 to 100 times larger than what would actually occur.

The use of conservative analyses is not an important problem or disadvantage in this EIS since all of the alternatives have been evaluated using the same methods and data, allowing a fair comparison of all of the alternatives on the same basis. Furthermore, even using these conservative analytical methods, the risks for all of the alternatives are small, which greatly reduces the significance of any uncertainty analysis parameters.

F.2 TOXIC CHEMICAL ISSUES AT NAVAL SPENT NUCLEAR FUEL EXAMINATION AND STORAGE SITES

The INEL-ECF is a large laboratory facility used to receive, examine, and ship naval nuclear fuel and irradiated test specimen assemblies. In order to accomplish these tasks, some chemicals classified as toxic are involved in a variety of operations and thus a potential exists for releases of toxic chemicals due to human error and failure or malfunctioning of equipment.

This section provides the results of an evaluation of both normal operations and accidents that could result in toxic chemical releases. This section describes how facilities and operations were selected for analysis, discusses the computer codes used in the analysis, presents the weather conditions and atmospheric dispersion, defines the hypothetical accidents which would produce the most severe consequences, and estimates the potential health effects. Each alternate location's specific population and meteorology were used to produce estimated consequences for each operation and accident.

F.2.1 Toxic Chemical Inventory

Some chemicals classified as toxic are routinely used in a variety of operations at the INEL-ECF. Table F.2-1 provides the INEL-ECF Chemical Inventory. This inventory was developed from the Naval Reactors Facility Superfund Amendments and Reauthorization Act (SARA) Section 312 chemical inventory (INEL 1993). Those chemicals specifically stored and used at INEL-ECF as well as those used for facility support (e.g., fuel oil, diesel fuel, sulfuric acid, and sodium hydroxide) were included. Chemicals at INEL-ECF that were (a) in excess of 500 pounds, or (b) in excess of reportable quantities (usually 1 pound) on the EPA Title III List of Lists (EPA 1992a) were evaluated. The chemicals in the EPA Title III List of Lists are the hazardous chemicals defined in:

- SARA Section 302 Extremely Hazardous Substances (CFR 1992a)
- CERCLA Hazardous Substances (CFR 1992b)
- SARA Section 313 Toxic Chemicals (CFR 1992c)
- RCRA Hazardous Wastes (CFR 1992d)
- EPA list of 100 extremely hazardous chemicals (FR 1993).

Table F.2-1. INEL-ECF chemical inventory.

CAS No.	Chemical Name	Weight Total (pounds)	Weight Unit[1] (pounds)
Chemicals Used for Water Pool Operations			
60-00-4	Ethylenediaminetetraacetic Acid (EDTA) (reagent for water analyses)	46.3	1.1
75-71-8	Dichlorodifluoromethane (CFC-12) (refrigerant in coolers for pool water)	30.0	30.0
Chemicals Used for Examination Operations			
60-29-7	Ethyl Ether	5.7	5.7
67-63-0	Isopropyl Alcohol	100.6	6.6
123-31-9	Hydroquinone (photographic film developer)	65.5	3.3
144-55-8	Sodium Bicarbonate	198.0	99.0
302-01-2	Hydrazine	3.7	1.8
7664-41-7	Ammonia[2]	2.8	0.28
7727-37-9	Diatomic Nitrogen	643	125
Chemicals Used for Facility Support			
107-21-1	Ethylene Glycol (anti-freeze and paint additive)	516.1	514.0
115-07-1	Propylene (Propene)	0.01	0.005
1310-73-2	Sodium Hydroxide (boiler water pH control)	43260	43260
7664-93-9	Sulfuric Acid (boiler and cooling tower water pH control)	96427	96427
68476-33-5	Fuel Oil #5	776210	204270
68476-34-6	Diesel Fuel #2	14316	10735
72623-83-7	Hydrotreated Lubricating Oil	882.6	413
Chemical Used for Nuclear Poison			
1332-77-0	Potassium Tetraborate	17000	10

[1] The quantities in this column represent the amount of chemical stored in the largest single container as identified in the INEL-ECF chemical inventory.
[2] The ammonia is present as ammonium hydroxide.

In order to evaluate the alternate locations, the same inventory of chemicals at the INEL-ECF was used at the DOE sites; namely, the Savannah River Site, the Hanford Site, the Nevada Test Site, and the Oak Ridge Reservation. In addition, the Barnwell Nuclear Fuel Plant (hereafter referred to as the Barnwell Plant), which is adjacent to the Savannah River Site, was evaluated along with the DOE sites. Since the shipyards would not be involved with examination operations (except for Puget Sound), of the chemicals listed, only diesel fuel would be available in a substantial quantity, in the form of fuel stored at the shipyards. Although several of the chemicals listed in Table F.2-1 are water treatment chemicals associated with water pool operations and small water pools may be needed at the shipyards for fuel storage and inspection, the shipyard would already have on-hand similar water treatment chemicals for other operations at the shipyard. Therefore, an increase in the quantities or types of chemicals at the shipyards was considered to be very small and thus did not require evaluation. In addition, even though the Kenneth A. Kesselring Site is not a shipyard, this facility would also not be involved with examination operations. Therefore, this facility was evaluated in the same manner as the shipyards.

F.2.2 Computer Modeling to Estimate Toxic Chemical Exposures

Factors such as locations of affected persons, terrain, meteorological conditions, release conditions, and characteristics of the chemical inventory are required as input parameters for calculations to determine human exposure from airborne releases of toxic chemicals. This section describes the computer models used to perform exposure estimates. Specific input parameters used in the analyses are summarized in the appropriate subsection for normal operations and accident conditions. The EPIcode was used to evaluate toxic chemical releases resulting from accidents, and the ISC2 code was used to evaluate releases from normal operations.

F.2.2.1 EPIcode™. The Emergency Prediction Information Computer Code (EPIcode™) is the computer code chosen for estimating airborne concentrations resulting from most releases of toxic chemicals (Homann 1988). Like RSAC, EPIcode uses the well-established Gaussian Plume Model to calculate the airborne toxic chemical concentrations usually at the same downwind locations as RSAC. The EPIcode library contains information on over 600 toxic substances listed by the American Conference of Governmental Industrial Hygienists in the EPIcode Manual. EPIcode also allows user description of substances not included in the library. A step-by-step flow chart of the main EPIcode features (up to the output options) is shown in Figure F.2-1.

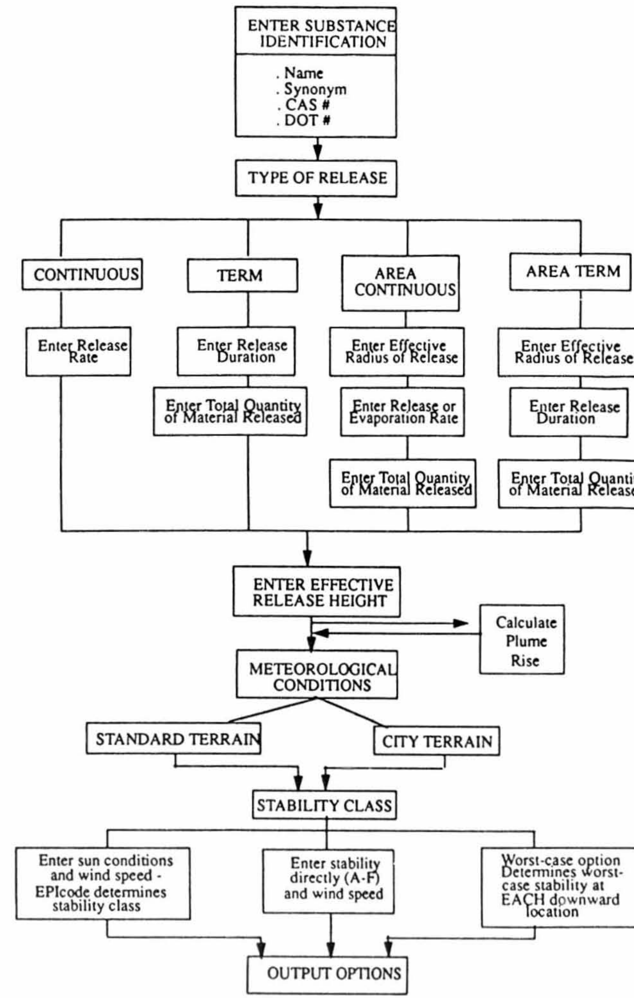

Figure F.2-1. Flow sheet for EPIcode (Homann 1988).

As shown in Figure F.2-1, the continuous release models require specification of the source term as an ambient concentration and a release rate. For releases over a specific time interval (i.e., term releases), the user specifies the release duration and the total quantity of material released.

Area continuous and area term releases are useful in calculating the effects of a release from pools of spilled volatile liquids. The user must enter the radius of the circle encompassing the spill area. Also entered is the temperature of the pool and ambient temperature to establish release rate from a liquid spill. An upwind virtual point source, which results in an initial lateral diffusion equal to the effective radius of the area source, is used to model an area release.

By specifying a release quantity, release duration, and release area, the user effectively proposes a release rate per unit spill area. The release quantity is defined as a source term (Q) or fraction of the material at risk. The concepts and defined terms are the same as for radiological calculations. EPIcode confirms that the volatility of the spilled substance can support such a release rate. If the proposed release rate exceeds the saturation conditions at the release temperature, EPIcode calculates a lower release rate and a corresponding longer release time.

In calculating effective release height, the actual plume height may not be the physical release height, e.g., the stack height. Plume rise can occur because of the velocity of a stack emission and the temperature differential between the stack effluent and the surrounding air. EPIcode calculates both the momentum plume rise and the buoyant plume rise and chooses the greater of the two results. Since this effective increase in release height leads to lower concentrations at the ground level, the physical release heights were used to calculate the concentrations that the general public may be exposed to during accidental releases of toxic substances. This approach will always yield conservative estimates.

In this application, the standard terrain calculation of EPIcode is always used. Downwind concentrations were calculated using both 95% and 50% meteorological conditions (Section F.1.3.5). The elevation of the affected person is always ground level (0 meters) and, as in RSAC-5, the mixing layer height is always 400 meters (1320 feet). The deposition velocities used (Section F.2.4.2.1.3) are somewhat different than those of RSAC-5, but they are still conservatively low.

As described in its user manual (Homann 1988), EPIcode also includes the following steps:

- Treating a release as instantaneous vs. continuous depending upon the plume length at the specific downwind location being considered
- Correcting the concentration for sampling time
- Adjusting the wind speed for release height
- Depleting the plume as a function of downwind distance
- Adjusting the standard deviations of the crosswind and vertical concentrations for brief releases.

As output, EPIcode can generate data plots of mean toxic chemical concentration (during a specified averaging time) as a function of downwind distance. From these graphs and numerical output, the concentrations for the worker at 100 meters (330 feet) (the shortest distance for which EPIcode calculates), for the nearest public access (NPA), for the maximum off-site individual (MOI), and for nearby communities are determined and evaluated for health effects.

EPIcode was selected as the computer code for release analysis of chemicals amenable to Gaussian modeling after comparison with a number of codes, primarily CHARM and ARCHIE. It was judged more applicable for this application than either the CHARM code or the comparable ARCHIE code.

F.2.2.2 ISC2 Code. The Industrial Source Complex (ISC2) model is a widely used, publicly available, and accepted EPA regulatory model which employs straight line (i.e., uniform wind field) Gaussian diffusion to estimate pollutant dispersion (EPA 1992b). ICS2 is an appropriate model for industrial complexes in rural or urban areas with transport distances less than 50 kilometers (30 miles). This model employs a standard meteorological data set requiring single point hourly wind speed, wind direction, ambient air temperature, atmospheric stability, and vertical mixing height values. Also, the ISC2 model is able to account for variations in pollutant concentrations due to the influence of nearby structures.

In addition to the ISC2 model, the MESOPUFF II model was also evaluated. MESOPUFF II is a regional (mesoscale) scale model that takes into account a varying wind field. Past trajectory analyses at the INEL have demonstrated that plumes may undergo many changes in direction due to the varying winds common to the INEL vicinity. The number of changes is partially dependent on release time and transport duration. The plume transport and estimation of pollutant concentration beyond 12 miles (20 kilometers) is best modeled using spatially varying wind data. Although not used as a basis for determining or enforcing compliance with regulations, it is used on a case-by-case basis. The model is also readily available to the public.

Upon review of the ISC2 and MESOPUFF II models, the decision was made to utilize ISC2 for the dispersion analysis of pollutants emitted from stationary sources. ISC2 is able to reasonably and accurately predict downwind pollutant concentrations within 30 miles (50 kilometers) by taking into account multiple point and area emission sources, evaluating hourly meteorological data, and determining the effects of nearby structures.

F.2.3 Health Effects

Toxic constituents dispersed during an accident could induce adverse health effects among exposed individuals. This possible impact is assessed by comparing the airborne concentrations of each substance at specified downwind locations to standard accident exposure guidelines for chemical toxicity.

Where available, Emergency Response Planning Guideline (ERPG) values are used for this comparison. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (Rusch 1993). ERPG values are specific for each substance, and are derived for each of three general severity levels:

- Exposure to concentrations greater than ERPG-1 values results in an unacceptable likelihood that one would experience mild transient adverse health effects, or perception of a clearly defined objectionable odor.

- Exposure to concentrations greater than ERPG-2 values results in an unacceptable likelihood that one would experience or develop irreversible or other serious health effects, or symptoms that could impair one's ability to take protective action.

- Exposure to concentrations greater than ERPG-3 values results in an unacceptable likelihood that one would experience or develop life-threatening health effects.

Where ERPG values have not been derived for a toxic substance, other chemical toxicity values are substituted, as follows:

- For ERPG-1, Threshold Limit Value, Time-Weighted Average (TLV-TWA) values (ACGIH 1993) are substituted: The TWA is the time-weighted average concentration for a normal 8-hour workday and a 40-hour workweek, to which nearly all workers may be repeatedly exposed, day after day, without adverse effect.

- For ERPG-2, Level of Concern values (equal to 0.1 of Immediately Dangerous to Life or Health) are substituted: Level of Concern is defined as the concentration of a hazardous substance in air, above which there may be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time (EPA 1987).

- For ERPG-3, Immediately Dangerous to Life or Health (IDLH) values are substituted: IDLH is defined as the maximum concentration from which a person could escape within 30 minutes without a respirator and without experiencing any effects which would impair the ability to escape or irreversible side effects (NIOSH 1990).

Possible health effects associated with exceeding an ERPG-2 or -3 value are specific for each substance of concern, and must be characterized in that context. When concentrations are found to exceed an ERPG or substitute value, the specific toxicological effects for the chemicals of concern are considered in describing possible health effects associated with exceeding a threshold value.

ERPG values are based upon a 1-hour exposure of a member of the general population. In this EIS, exposures resulting from the release of toxic chemicals during an accident condition were postulated to occur over a period of 1 hour or less to allow for a direct comparison to the ERPG values. This approach provides an additional element of conservatism in the evaluation of accidents with releases that last much less than 1 hour.

In addition to comparing the airborne concentrations of each substance to standard accident exposure guidelines, each substance was evaluated to determine if it has the potential for future carcinogenic health impacts. If a particular substance has this potential, the Integrated Risk Information System (IRIS) (TOXnet 1993) was reviewed and if sufficient toxicological information was available, a future potential likelihood of developing cancer was determined. If sufficient information from IRIS was not available, alternative evaluation methods, including comparison to ambient air quality criteria, were substituted.

The impact of normal operations was also evaluated. This impact was assessed by comparing the airborne concentrations of each substance at specified downwind locations to the National Ambient Air Quality Standards (NAAQS) assigned for each substance. NAAQS consist of national primary and secondary ambient air quality standards (CFR 1991). National primary ambient air quality standards define levels of air quality which the EPA judges are necessary, with an adequate margin of safety, to protect the public health. National secondary ambient air quality standards define levels of air quality which the EPA judges are necessary to protect the public welfare from any known or anticipated adverse effects of a pollutant. As a result, the immediate as well as cumulative impact of normal operations was evaluated by comparing the airborne concentrations of each substance to the NAAQS.

F.2.4 Analysis Description and Results

The analysis results for both normal operations and accident conditions are reported for each location analyzed. Detailed estimated concentrations and ERPG levels, expressed in milligrams per cubic meter (mg/m^3), are reported in tabular form for a worker, maximally exposed collocated worker (MCW), maximally exposed off-site individual (MOI), and maximally exposed individual at the nearest public access (NPA). A complete description of these individuals is provided in Section F.1.3.2.

F.2.4.1 Normal Operations.

F.2.4.1.1 Source of Emissions. Emissions resulting from normal operations involving toxic chemicals listed in Table F.2-1 were evaluated. It was determined that the burning of Number 5 fuel oil in the facility's boilers and the burning of Number 2 diesel fuel in the facility's emergency diesel generators represented the largest sources of emissions under normal operations and thus provide the conditions producing the most severe consequences for evaluation. These normal operations result in the release of oxides of nitrogen (90% nitric oxide and 10% nitrogen dioxide), sulfur dioxide, particulates (PM-10), lead, and volatile organic compounds (VOCs). The airborne release of these chemicals was evaluated for effects on the on-site workers, MCW, NPA, and MOI.

The emissions that occur due to normal operations at the INEL-ECF were evaluated using the ISC2 code. These releases were also used at the alternate locations (Hanford, Savannah River, Nevada Test Site, Barnwell Plant, and Oak Ridge) for evaluation purposes. Heating boilers and emergency diesel generators already exist at the alternate shipyard locations and thus selection of these alternate locations would not result in a measurable increase in emissions. Therefore, routine releases from shipyard locations were not considered.

F.2.4.1.2 Conditions and Key Parameters.

- Number 5 fuel oil was burned in facility boilers for space heating.
- Number 2 diesel fuel was burned in facility emergency diesel generators.
- Source term was based on the INEL report on routine yearly releases (NRF 1993) which included:
    - 1.02 tons per year of carbon monoxide released
    - 9.04 tons per year of oxides of nitrogen released
    - 33.7 tons per year of sulfur dioxide
    - 1.54 tons per year of particulates

- 5.86×10^{-4} tons per year of lead
- 0.18 tons per year of volatile organic compounds.

- Forty percent of the total boiler and emergency diesel generator use for the Naval Reactors Facility was attributed to the INEL-ECF.
- Three point sources (one representing boilers and two representing emergency diesel generators) were used.
- Stack diameters of 1.07 meters (3.5 feet) for boilers and 0.305 meter (1 foot) for emergency diesel generators were used.
- Stack gas exit velocities of 21.8 meters per second (72 feet per second) for boilers and 44.2 meters per second (145 feet per second) for emergency diesel generators were used.
- Stack gas exit temperatures of 505°K for boilers and 794°K for emergency diesel generators were used.
- Worker concentrations were based on 16 sector polar grids. Other affected locations were defined as discrete points.
- DOE site meteorological data were used for evaluations at the Naval Reactors Facility, Hanford, Nevada Test Site, and Oak Ridge. Meteorological data from the closest National Weather Service Station were used for evaluations at Savannah River and the Barnwell Plant.

F.2.4.1.3 Results. The airborne concentrations, averaged over the duration of each exposure, were calculated by ISC2 for the worker, MCW, NPA, and MOI using normal meteorology. Tables F.2.4.1-1 through -6 list the downwind concentrations at various locations. The airborne concentrations were compared to respective NAAQS values where available. The NAAQS are as follows:

Carbon monoxide. The national primary ambient air quality standards for carbon monoxide are 10 mg/m^3 for an 8-hour average concentration not to be exceeded more than once per year, and 40 mg/m^3 for a 1-hour average concentration not to be exceeded more than once per year.

Sulfur oxides. The national primary ambient air quality standards for sulfur oxides that are measured as sulfur dioxide are 0.08 mg/m^3 as an annual arithmetic mean and 0.365 mg/m^3 as a maximum 24-hour concentration not to be exceeded more than once per year. The national secondary ambient air quality standards are 1.3 mg/m^3 as a maximum 3-hour concentration not to be exceeded more than once per year.

Nitrogen dioxide. The national primary and secondary ambient air quality standard for nitrogen dioxide is 0.1 mg/m^3 as an annual arithmetic mean.

Lead. The national primary and secondary ambient air quality standard for lead and its compounds that are measured as elemental lead is 1.5×10^{-3} mg/m^3 as a maximum arithmetic mean averaged over a calendar quarter.

Particulate matter. The national primary and secondary ambient air quality standard for particulate matter is 0.05 mg/m^3 as an annual arithmetic mean and 0.15 mg/m^3 as a maximum 24-hour concentration.

A comparison of the downwind concentrations provided in Tables F.2.4.1-1 through -6 with the NAAQS identified above indicates that no NAAQS is exceeded for normal operations.

Table F.2.4.1-1. Summary of chemical concentrations for normal operations at the INEL Expended Core Facility.

	CHEMICAL CONCENTRATIONS mg/m^3						
	Carbon Monoxide	Sulfur Dioxide	Nitric Oxide	Nitrogen Dioxide	Lead	VOC	PM-10
Worker	4.6×10^{-5}	5.5×10^{-4}	1.9×10^{-4}	2.1×10^{-5}	9.0×10^{-9}	1.9×10^{-5}	2.7×10^{-5}
MCW	3.7×10^{-6}	9.5×10^{-5}	2.6×10^{-5}	2.9×10^{-6}	2.0×10^{-9}	8.5×10^{-7}	4.6×10^{-6}
MOI	7.7×10^{-7}	2.3×10^{-5}	5.8×10^{-6}	6.4×10^{-7}	$< 1.0 \times 10^{-9}$	1.6×10^{-7}	1.1×10^{-6}
NPA	7.7×10^{-7}	2.3×10^{-5}	5.8×10^{-6}	6.4×10^{-7}	$< 1.0 \times 10^{-9}$	1.6×10^{-7}	1.1×10^{-6}

Table F.2.4.1-2. Summary of chemical concentrations for normal operations at Hanford.

	CHEMICAL CONCENTRATIONS mg/m^3						
	Carbon Monoxide	Sulfur Dioxide	Nitric Oxide	Nitrogen Dioxide	Lead	VOC	PM-10
Worker	2.9×10^{-5}	1.5×10^{-4}	1.3×10^{-4}	1.0×10^{-5}	3.0×10^{-9}	1.1×10^{-5}	1.4×10^{-5}
MCW	1.6×10^{-5}	2.1×10^{-4}	9.6×10^{-5}	1.1×10^{-5}	5.0×10^{-9}	4.7×10^{-6}	1.5×10^{-5}
MOI (New ECF)*	1.0×10^{-6}	3.2×10^{-5}	8.0×10^{-6}	8.9×10^{-7}	1.0×10^{-9}	2.0×10^{-7}	1.5×10^{-6}
MOI (FMEF)**	1.4×10^{-6}	4.0×10^{-5}	1.1×10^{-5}	1.2×10^{-6}	1.0×10^{-9}	3.0×10^{-7}	1.9×10^{-6}
NPA	1.3×10^{-6}	4.1×10^{-5}	1.0×10^{-5}	1.1×10^{-6}	1.0×10^{-9}	2.6×10^{-7}	1.9×10^{-6}

*MOI (New ECF) applies if spent fuel facility is constructed at the 200 Area on the Hanford Site.
**MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Materials Examination Facility.

Table F.2.4.1-3. Summary of chemical concentrations for normal operations at Savannah River.

	CHEMICAL CONCENTRATIONS mg/m^3						
	Carbon Monoxide	Sulfur Dioxide	Nitric Oxide	Nitrogen Dioxide	Lead	VOC	PM-10
Worker	1.5×10^{-5}	6.4×10^{-5}	6.4×10^{-5}	7.1×10^{-6}	1.0×10^{-9}	6.2×10^{-6}	5.9×10^{-6}
MCW	9.4×10^{-6}	1.6×10^{-4}	5.7×10^{-5}	6.3×10^{-6}	3.0×10^{-9}	2.8×10^{-6}	8.7×10^{-6}
MOI	1.8×10^{-6}	4.8×10^{-5}	1.3×10^{-5}	1.4×10^{-6}	1.0×10^{-9}	3.8×10^{-7}	2.3×10^{-6}
NPA	8.6×10^{-7}	2.4×10^{-5}	6.3×10^{-6}	7.0×10^{-7}	$< 1.0 \times 10^{-9}$	1.9×10^{-7}	1.1×10^{-6}

393

Table F.2.4.1-4. Summary of chemical concentrations for normal operations at the Nevada Test Site.

	CHEMICAL CONCENTRATIONS mg/m^3						
	Carbon Monoxide	Sulfur Dioxide	Nitric Oxide	Nitrogen Dioxide	Lead	VOC	PM-10
Worker	9.0×10^{-5}	3.6×10^{-4}	4.0×10^{-4}	4.5×10^{-5}	7.0×10^{-9}	3.8×10^{-5}	4.1×10^{-5}
MCW	2.5×10^{-7}	7.3×10^{-6}	1.9×10^{-6}	2.1×10^{-7}	$< 1.0 \times 10^{-9}$	5.2×10^{-8}	3.5×10^{-7}
MOI	7.9×10^{-7}	2.3×10^{-5}	5.9×10^{-6}	6.6×10^{-7}	$< 1.0 \times 10^{-9}$	1.6×10^{-7}	1.1×10^{-6}

Table F.2.4.1-5. Summary of chemical concentrations for normal operations at Oak Ridge.

	CHEMICAL CONCENTRATIONS mg/m^3						
	Carbon Monoxide	Sulfur Dioxide	Nitric Oxide	Nitrogen Dioxide	Lead	VOC	PM-10
Worker	6.4×10^{-5}	3.0×10^{-4}	2.8×10^{-4}	3.1×10^{-5}	5.0×10^{-9}	2.6×10^{-5}	2.7×10^{-5}
MCW	1.6×10^{-6}	2.6×10^{-5}	9.6×10^{-6}	1.1×10^{-6}	$< 1.0 \times 10^{-9}$	5.0×10^{-7}	1.5×10^{-6}
MOI	1.4×10^{-5}	2.5×10^{-4}	8.8×10^{-5}	9.8×10^{-6}	4.0×10^{-9}	4.3×10^{-6}	1.4×10^{-5}
NPA	1.9×10^{-5}	3.1×10^{-4}	1.1×10^{-4}	1.2×10^{-5}	5.0×10^{-9}	5.6×10^{-6}	1.7×10^{-5}

Table F.2.4.1-6. Summary of chemical concentrations for normal operations at the Barnwell Plant.

	CHEMICAL CONCENTRATIONS mg/m^3						
	Carbon Monoxide	Sulfur Dioxide	Nitric Oxide	Nitrogen Dioxide	Lead	VOC	PM-10
Worker	1.5×10^{-5}	6.5×10^{-5}	6.4×10^{-5}	7.1×10^{-6}	1.0×10^{-9}	6.2×10^{-6}	5.9×10^{-6}
MCW	1.9×10^{-6}	4.7×10^{-5}	1.3×10^{-5}	1.5×10^{-6}	1.0×10^{-9}	4.5×10^{-7}	2.3×10^{-6}
MOI	5.9×10^{-6}	1.4×10^{-4}	4.0×10^{-5}	4.5×10^{-6}	2.0×10^{-9}	1.5×10^{-6}	7.0×10^{-6}
NPA	5.9×10^{-6}	1.4×10^{-4}	4.0×10^{-5}	4.5×10^{-6}	2.0×10^{-9}	1.5×10^{-6}	7.0×10^{-6}

394

F.2.4.2 Accidents. Spillage of chemicals with a subsequent fire was evaluated for the bounding accident involving toxic chemicals. The toxic chemicals that could be involved in the postulated accident are described in Section F.2.1. As was noted in that section, the extensive listing of chemicals provided in Table F.2-1 would be applicable only at sites involved with fuel examination. The bounding accident evaluated for spent nuclear fuel storage in water pools at shipyard locations was a diesel fuel spill and fire. A diesel fuel fire involving spent nuclear fuel shipping containers aboard a ship at sea in Puget Sound was also evaluated.

Evaluation of the chemical spill with fire accident (excluding diesel fuel) at the alternate sites (INEL-ECF, Hanford, Savannah River, Nevada Test Site, Oak Ridge, and the Barnwell Plant) where naval spent nuclear fuel examinations may be conducted is presented in Section F.2.4.2.1. Evaluation of diesel fuel fires at shipyards and aboard ship in Puget Sound, as well as at INEL-ECF, Hanford, Savannah River, Nevada Test Site, Barnwell Plant, and Oak Ridge, is described in Section F.2.4.2.2.

These accidents incorporate spillage of the entire amount of a given chemical accompanied by a fire. The initiating event might be, for example, an airplane crash or ship collision. Such an accident bounds simpler chemical spills, such as handling accidents involving limited or unit (see Table F.2-1) amounts of a chemical, which were also considered. Consequently, only results for the fire accident are provided. The analyses utilize meteorological (see Section F.1.3.5) and demographic parameters specific to the evaluated location.

The toxic chemicals evaluated in the accident analyses would be used and stored in a number of different areas within the facility. Fuel oils, sulfuric acid, and sodium hydroxide would be expected to be located outside facility buildings in storage tanks. Other chemicals used for facility support and operation would likely be stored in a variety of locations within facility buildings such as tool rooms, laboratories, craft shops, equipment rooms, chemical mixing areas, hot cells, and flammable cabinets. The probability of releasing all or most of these chemicals in a single accident such as an airplane crash would be quite low, less than 10^{-7} per year, as supported in Section F.3.5. However, the probability of releasing an individual or limited number of chemicals is expected to be greater than this level and include a consideration of storage locations, types, sizes, and numbers of containers, and types and frequencies of initiating events. For accidents that could result in a toxic chemical release, a probability of 5×10^{-3} per year (Ganti and Krasner 1984) was considered to be a reasonable upper level. This level was based on the probability that a structurally damaging industrial fire could occur.

F.2.4.2.1 Chemical Spill and Fire.

F.2.4.2.1.1 Accident Description. An accident might occur which caused toxic chemicals to spill, dispersed powdered toxic chemicals, and accelerated the vaporization of the toxic chemicals with a subsequent fire. The airborne release resulting from the involvement of the entire available amount of the toxic chemicals was evaluated with respect to the on-site workers, MCW, NPA, and MOI.

F.2.4.2.1.2 Source Term. The toxic chemicals involved in this hypothetical accident are provided in Table F.2-1. The entire amount of the toxic chemical might be involved due to the catastrophic nature of this accident.

F.2.4.2.1.3 Conditions and Key Parameters.

(1) Gases

- 100% of the gas was released to the atmosphere.
- Release period was 10 minutes.
- Release was a point source.
- Deposition velocity was 0.1 centimeter per second.

(2) Liquids

- 100% of the liquid was released to the atmosphere.
- The liquid was released into a pool of 0.1-inch depth.
- The liquid was at its boiling point.
- The release period was the longer of the calculated evaporation time or 10 minutes.
- Release area was equal to the pool area.
- Deposition velocity was 0.1 centimeter per second.

(3) Solids

- 1% of the solid was dispersed into the atmosphere as PM-10.
- Release period was 10 minutes.

- Release was a point source.
- Deposition velocity was 1.0 centimeter per second.

(4) Specific Chemicals

- CFC-12 could break down at elevated temperatures into hydrochloric acid (10%) and phosgene (1%) with the remaining (89%) released as CFC-12.
- The hypothetical sulfuric acid spill would be contained by a berm resulting in a pool release area of 443.2 square feet.
- The hypothetical spill of sodium hydroxide was in the form of an aqueous solution and was contained by a berm resulting in a pool release area of 374 square feet. A 10-minute period was used for this release, and the sodium hydroxide was dispersed as a particulate.

(5) Meteorology

- Wind speeds and atmospheric stability classifications used for the calculations were based on both 50% and 95% meteorology (Section F.1.3.5) to estimate downwind concentrations. The 95% meteorology included atmospheric stability classes A through F and wind speeds from 1.1 to 30 miles per hour.

(6) General

- Standard rural terrain was used since this most closely resembles the sites being evaluated.
- Release was calculated to occur at ground level.
- No evacuation of downwind populations was included, in order to obtain maximum estimates of effects; therefore, exposures were not reduced to account for this action.
- No credit was taken for building containment or filtration.
- Biological effects of exposure to each chemical were treated separately. This was done to account for a lack of a current methodology to evaluate the effects resulting from simultaneous multiple chemical exposures.

- To determine health impacts, the estimated concentrations were compared against the Emergency Response Planning Guidelines (ERPG) levels 1, 2, and 3 concentration limits or alternates.
- To determine the likelihood of developing cancer from exposure to hydrazine, a slope factor of 1.7 x 10^1 per mg/kg-day obtained from IRIS (TOXnet 1993) was used. In addition, the exposure time was based on the duration of the release, and individual breathing rates and sizes were the same as those used in Section F.1 for radiological accident evaluations using the Radiological Safety Analysis Computer Program (RSAC-5) (Wenzel 1993).

F.2.4.2.1.4 Results. The airborne concentrations, averaged over the duration of each exposure, were calculated using EPIcode for the alternate locations for the worker, MCW, NPA, and MOI for both 50% and 95% meteorology. The airborne concentrations were compared to respective ERPG values where available. However, ERPG values have not been derived for some of the chemicals. The effects of these substances were assessed by comparison with other appropriate values for toxic effects as discussed in Section F.2.3.3.

Tables F.2.4.2-1 through -12 list the downwind concentrations at various locations and corresponding ERPG values (or equivalent if TLV-TWA and IDLH concentrations are available). Hydrochloric acid and phosgene, from decomposition of CFC-12, sulfuric acid, and sodium hydroxide dominate the toxic chemical effects for on-site personnel. Concentrations of these chemicals above ERPG-3 levels might result in life-threatening effects. However, in no case is an ERPG-3 level exceeded for any member of the general public except for Oak Ridge where sulfuric acid concentrations could exceed ERPG-3 levels under both 50% and 95% meteorological conditions and sodium hydroxide concentrations could exceed ERPG-3 levels under 95% meteorological conditions. For the on-site workers, collocated workers, and any member of the general public that could be exposed to toxic chemicals at levels above ERPG-3, it is expected that actual toxic chemical exposures would be much less due to the mitigative measures that would be implemented (Section F.2.4.3).

Additional information on the toxic properties for the chemicals that dominate the toxic effects is provided below.

Hydrochloric acid is a irritant to the respiratory tract, skin, eyes, and mucous membranes. More severe exposures result in pulmonary edema, and often laryngeal spasm. A concentration of 53 mg/m^3 causes irritation of the throat after short exposure. Concentrations of 75-150 mg/m^3 are tolerable for 1 hour; concentrations of 1,500-3,000 mg/m^3 are dangerous, even for brief exposures (TOXnet 1993).

Phosgene, also known as carbonyl chloride, is a highly toxic, corrosive liquid with a low boiling point. It is toxic from intakes by inhalation, ingestion, and dermal absorption. Effects from exposure may include contact burns to the skin and eyes, shortness of breath, chest pain, severe pulmonary edema, and death. At low vapor concentrations, it smells like musty hay. At higher concentrations, it has a sharp and pungent odor. It is a severe irritant to the eyes and respiratory tract and can be fatal if inhaled, even for short durations and at low concentrations. Exposure to 12 mg/cm^3 can result in immediate irritation of the respiratory tract. 80 mg/m^3 may cause lung injuries within 2 minutes; 100 mg/m^3 for as little as 30 minutes is very dangerous; and 360 mg/m^3 is rapidly fatal for exposures of 30 minutes or less (TOXnet 1993).

Sulfuric acid mist can be strongly irritating to the skin, eyes, mucous membranes, and respiratory tract. Odor may be detected at concentrations of 1 mg/m^3; irritating effects may occur at concentrations of 1.1 mg/m^3. Inhalation of concentrations near 3 mg/m^3 may cause constriction of the air passage and choking sensations. At higher concentrations and durations of exposure, inhalation can cause pulmonary edema, emphysema, and permanent changes in pulmonary function (TOXnet 1993).

Sodium hydroxide dust can be irritating to the upper respiratory system. Irritating effects may occur at concentrations of 2 mg/m^3. At higher concentrations and durations of exposure, inhalation can cause extreme irritation of the respiratory tract and permanent changes in pulmonary function (TOXnet 1993).

Table F.2.4.2-1. Summary of chemical concentrations for chemical spill and fire at the INEL Expended Core Facility.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	3300	49	890	38	400	45	4.5	2300	6.4	2300
MCW	2.3	1.6×10^{-2}	0.45	1.2×10^{-2}	0.12	1.3×10^{-2}	1.3×10^{-3}	1.4	9.3×10^{-4}	0.60
MOI	1.5	1.0×10^{-2}	0.29	7.9×10^{-3}	7.7×10^{-2}	8.5×10^{-3}	8.5×10^{-4}	0.86	5.9×10^{-4}	0.39
NPA	1.6	1.1×10^{-2}	0.30	8.3×10^{-3}	8.1×10^{-2}	9.0×10^{-3}	9.0×10^{-4}	0.91	5.9×10^{-4}	0.39

Table F.2.4.2-2. Summary of chemical concentrations for chemical spill and fire at the INEL Expended Core Facility.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	4400	58	2200	150	1600	180	18	2800	7.7	2700
MCW	7.6	4.8×10^{-2}	2.6	8.3×10^{-2}	0.80	8.9×10^{-2}	8.9×10^{-3}	3.9	2.2×10^{-3}	1.5
MOI	3.6	2.3×10^{-2}	1.1	3.2×10^{-2}	0.30	3.4×10^{-2}	3.4×10^{-3}	1.9	8.8×10^{-4}	0.58
NPA	3.6	2.3×10^{-2}	1.1	3.2×10^{-2}	0.30	3.4×10^{-2}	3.4×10^{-3}	1.9	8.8×10^{-4}	0.58

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels could not be determined.

400

Table F.2.4.2-3. Summary of chemical concentrations for chemical spill and fire at Savannah River.

	CHEMICAL CONCENTRATIONS mg/m³ - 50% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	1500	19	370	14	150	16	1.6	1000	2.9	1200
MCW	32	0.25	6.6	0.19	1.9	0.21	2.1×10^{-2}	20	3.6×10^{-2}	22
MOI	1.3	8.7×10^{-3}	0.24	6.7×10^{-3}	6.4×10^{-2}	7.2×10^{-3}	7.2×10^{-4}	0.88	7.2×10^{-4}	0.47
NPA	1.3	8.7×10^{-3}	0.24	6.7×10^{-3}	6.4×10^{-2}	7.2×10^{-3}	7.2×10^{-4}	0.88	7.2×10^{-4}	0.47

Table F.2.4.2-4. Summary of chemical concentrations for chemical spill and fire at Savannah River.

	CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	4400	58	2200	150	1600	180	18	2800	7.7	2700
MCW	220	1.6	85	4.0	39	4.3	0.43	120	0.12	72
MOI	4.9	3.0×10^{-2}	1.6	4.7×10^{-2}	0.44	4.9×10^{-2}	4.9×10^{-3}	2.5	1.3×10^{-3}	0.85
NPA	4.9	3.0×10^{-2}	1.6	4.7×10^{-2}	0.44	4.9×10^{-2}	4.9×10^{-3}	2.5	1.3×10^{-3}	0.85

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels could not be determined.

Table F.2.4.2-5. Summary of chemical concentrations for chemical spill and fire at Hanford.

	CHEMICAL CONCENTRATIONS mg/m³ - 50% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	1500	19	370	14	150	16	1.6	1000	2.9	1200
MCW	46	0.36	9.6	0.28	2.7	0.30	3.0×10^{-2}	28	4.1×10^{-2}	26
MOI (New ECF)**	0.73	5.1×10^{-3}	8.1×10^{-2}	3.9×10^{-3}	3.8×10^{-2}	4.2×10^{-3}	4.2×10^{-4}	0.44	2.3×10^{-4}	0.16
MOI (FMEF)***	0.97	7.1×10^{-3}	0.19	5.4×10^{-3}	5.2×10^{-2}	5.8×10^{-3}	5.8×10^{-4}	0.96	7.8×10^{-4}	0.51
NPA	1.5	9.9×10^{-3}	0.29	7.9×10^{-3}	7.6×10^{-2}	8.5×10^{-3}	8.5×10^{-4}	0.86	7.3×10^{-4}	0.49

Table F.2.4.2-6. Summary of chemical concentrations for chemical spill and fire at Hanford.

	CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	4400	58	2200	150	1600	180	18	2800	7.7	2700
MCW	150	1.1	55	2.5	24	2.7	0.27	78	7.6×10^{-2}	45
MOI (New ECF)**	2.1	1.3×10^{-2}	0.47	1.3×10^{-2}	0.13	1.4×10^{-2}	1.4×10^{-3}	1.1	4.1×10^{-4}	0.28
MOI (FMEF)***	5.5	3.5×10^{-2}	1.8	5.4×10^{-2}	0.51	5.7×10^{-2}	5.7×10^{-3}	2.8	1.5×10^{-3}	0.99
NPA	5.3	3.3×10^{-2}	1.7	5.1×10^{-2}	0.48	5.4×10^{-2}	5.4×10^{-3}	2.7	1.4×10^{-3}	0.94

* IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels could not be determined.
**MOI (New ECF) applies if spent fuel facility is constructed at the 200 Area on the Hanford Site.
***MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Materials Examination Facility.

400

Table F.2.4.2-7. Summary of chemical concentrations for chemical spill and fire at the Nevada Test Site.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	530	6.8	130	5.1	53	5.9	0.59	820	1.2	490
MCW	0.22	1.5×10^{-3}	4.1×10^{-2}	1.1×10^{-3}	1.1×10^{-2}	1.2×10^{-3}	1.2×10^{-4}	0.12	2.1×10^{-4}	0.14
MOI	0.74	5.4×10^{-3}	0.14	4.0×10^{-3}	3.8×10^{-2}	4.4×10^{-3}	4.4×10^{-4}	0.97	7.0×10^{-4}	0.46

Table F.2.4.2-8. Summary of chemical concentrations for chemical spill and fire at the Nevada Test Site.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	4400	58	2200	150	1600	180	18	2800	7.7	2700
MCW	5.9	3.7×10^{-2}	1.9	5.8×10^{-2}	0.55	6.2×10^{-2}	6.2×10^{-3}	3.0	1.6×10^{-3}	1.1
MOI	7.3	4.6×10^{-2}	2.5	7.8×10^{-2}	0.76	8.4×10^{-2}	8.4×10^{-3}	3.8	2.2×10^{-3}	1.4

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels could not be determined.

403

Table F.2.4.2-9. Summary of chemical concentrations for chemical spill and fire at Oak Ridge.

	CHEMICAL CONCENTRATIONS mg/m³ - 50% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	3300	49	890	38	400	45	4.5	2300	6.4	2300
MCW	34	0.27	7.1	0.21	2.0	0.22	2.2×10^{-2}	21	3.0×10^{-2}	19
MOI	310	2.8	68	2.1	21	2.4	0.24	190	0.38	210
NPA	440	4.3	100	3.2	32	3.7	0.37	280	0.60	310

Table F.2.4.2-10. Summary of chemical concentrations for chemical spill and fire at Oak Ridge.

	CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	4400	58	2200	150	1600	180	18	2800	7.7	2700
MCW	110	0.75	41	1.9	18	2.0	0.20	58	5.4×10^{-2}	32
MOI	930	8.4	400	22	220	24	2.4	540	0.82	410
NPA	1300	13	590	33	340	38	3.8	790	1.3	630

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels could not be determined.

404

Table F.2.4.2-11. Summary of chemical concentrations for chemical spill and fire at the Barnwell Plant.

	CHEMICAL CONCENTRATIONS mg/m³ - 50% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	1500	19	370	14	150	16	1.6	1000	2.9	1200
MCW	0.89	6.4×10^{-3}	0.17	4.9×10^{-3}	4.6×10^{-2}	5.2×10^{-3}	5.2×10^{-4}	0.83	9.0×10^{-4}	0.59
MOI	6.1	4.3×10^{-2}	1.2	3.4×10^{-2}	0.32	3.6×10^{-2}	3.6×10^{-3}	4.9	4.9×10^{-3}	3.2
NPA	6.1	4.3×10^{-2}	1.2	3.4×10^{-2}	0.32	3.6×10^{-2}	3.6×10^{-3}	4.9	4.9×10^{-3}	3.2

Table F.2.4.2-12. Summary of chemical concentrations for chemical spill and fire at the Barnwell Plant.

	CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOROLOGY									
	Ethylene Glycol ERPG-1 127 ERPG-2 * ERPG-3 *	Hydrazine ERPG-1 0.13 ERPG-2 10 ERPG-3 100	Isopropyl Alcohol ERPG-1 983 ERPG-2 2950 ERPG-3 29500	Ammonia ERPG-1 18 ERPG-2 140 ERPG-3 700	CFC-12 ERPG-1 4950 ERPG-2 24750 ERPG-3 247500	Hydrochloric Acid ERPG-1 4.5 ERPG-2 30 ERPG-3 150	Phosgene ERPG-1 0.4 ERPG-2 0.8 ERPG-3 4.0	Sulfuric Acid ERPG-1 2 ERPG-2 10 ERPG-3 30	Hydroquinone ERPG-1 2 ERPG-2 * ERPG-3 *	Sodium Hydroxide ERPG-1 2 ERPG-2 25 ERPG-3 250
Worker	4400	58	2200	150	1600	180	18	2800	7.7	2700
MCW	11	6.4×10^{-2}	3.5	0.13	1.3	0.14	1.4×10^{-2}	5.4	3.2×10^{-3}	2.0
MOI	28	0.18	10	0.41	3.9	0.44	4.4×10^{-2}	15	1.1×10^{-2}	6.9
NPA	28	0.18	10	0.41	3.9	0.44	4.4×10^{-2}	15	1.1×10^{-2}	6.9

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels could not be determined.

405

In addition to comparing the airborne concentrations to their respective ERPG or other appropriate values, each substance was evaluated to determine if it has the potential for future carcinogenic health impacts. It was determined that exposure to hydrazine could result in an increased likelihood for developing cancer. Tables F.2.4.2-13 and F.2.4.2-14 provide the future potential likelihood for developing cancer from exposure to hydrazine for the worker, MCW, and MOI at the alternate locations under 50% and 95% meteorological conditions, respectively.

Table F.2.4.2-13. Future potential likelihood for developing cancer from hydrazine - 50% meteorology.

	INEL Expended Core Facility	Savannah River	Hanford*	Nevada Test Site	Oak Ridge	Barnwell Plant
Worker	9.3×10^{-5}	3.6×10^{-5}	3.6×10^{-5}	1.3×10^{-5}	9.3×10^{-5}	3.6×10^{-5}
MCW	3.0×10^{-8}	4.8×10^{-7}	6.8×10^{-7}	2.8×10^{-9}	5.1×10^{-7}	1.2×10^{-8}
MOI	1.5×10^{-8}	1.3×10^{-8}	7.6×10^{-9}	8.1×10^{-9}	4.2×10^{-6}	6.4×10^{-8}

Table F.2.4.2-14. Future potential likelihood for developing cancer from hydrazine - 95% meteorology.

	INEL Expended Core Facility	Savannah River	Hanford*	Nevada Test Site	Oak Ridge	Barnwell Plant
Worker	3.8×10^{-4}	3.8×10^{-4}	3.8×10^{-4}	3.8×10^{-4}	3.8×10^{-4}	3.8×10^{-4}
MCW	2.0×10^{-7}	6.7×10^{-6}	4.6×10^{-6}	1.6×10^{-7}	3.2×10^{-6}	2.7×10^{-7}
MOI	7.8×10^{-8}	1.0×10^{-7}	4.4×10^{-8}	1.6×10^{-7}	2.9×10^{-5}	6.1×10^{-7}

* MOI shown applies to new ECF if spent fuel facility is constructed at the 200 Area on the Hanford Site. A future potential carcinogenic risk of 1.1×10^{-8} (50% meteorology) and 1.2×10^{-7} (95% meteorology) applies to a spent fuel facility constructed at the Fuels and Materials Examination Facility.

F.2.4.2.2 Fire Involving Diesel Fuel.

F.2.4.2.2.1 Accident Description. A catastrophic failure of the diesel fuel storage tank facility was postulated to occur. This could result in the spilling of the entire quantity of diesel fuel and a subsequent fire. The airborne release of toxic chemicals resulting from the fire was evaluated with respect to the on-site workers, MCW, NPA, and MOI as applicable for the accident site.

F.2.4.2.2.2 Source Term. The material involved in this accident was diesel fuel with the fire generating the following toxic chemicals due to combustion:

- Carbon monoxide
- Oxides of nitrogen (90% nitric oxide and 10% nitrogen dioxide)
- Lead
- Sulfur dioxide.

F.2.4.2.2.3 Conditions and Key Parameters.

- For alternate DOE sites and the Barnwell Plant, the diesel fuel was stored in bulk storage tanks.
- For shipyards, the diesel fuel was stored in a portable diesel power unit.
- For the ship accident, the diesel fuel was stored in large tanks adjacent to the hold.
- For alternate DOE sites and the Barnwell Plant, 1950 gallons of diesel fuel could be spilled.
- For shipyards, 315 gallons of diesel fuel could be spilled.
- For the ship accident, 121,000 gallons of diesel fuel could be spilled.
- For all facilities, the entire quantity of diesel fuel was spilled and ignited in open air.
- For alternate DOE sites and the Barnwell Plant, the spill area was 261 square feet.
- For shipyards, the spill area was 66 square feet.
- For the ship accident, the spill area used was 4812 square feet.
- For alternate DOE sites and the Barnwell Plant, the entire amount of diesel fuel was consumed by the fire over a 2-hour period.
- For shipyards, the entire amount of diesel fuel was consumed by the fire over a 1-hour period.

- For the ship accident, the entire amount of diesel fuel was consumed by the fire over a 6-hour period.
- For all facilities, the releases per gallon of fuel burned were as follows:
  Carbon monoxide = 0.34 pound
  Oxides of nitrogen = 1.58 pounds
  Lead = 4.2×10^{-6} pound
  Sulfur dioxide = 0.105 pound.
- For alternate DOE sites, the Barnwell Plant, and shipyards, the airborne release of toxic chemicals occurred at ground level.
- For the ship accident, the airborne release of toxic chemicals occurred at 48 feet above the sea (i.e., at the middle of the flame height above the cargo hatch) for evaluation of land-based exposures. For shipboard exposures, a release height of zero was used.
- For all facilities, standard rural terrain was used and building wake effects were not considered.
- For all facilities, wind speeds and atmospheric stability classifications were based on both 50% and 95% meteorology (Section F.1.3.5).
- For all facilities, no evacuation of downwind populations occurred and the biological effects of chemical exposure act uniquely and do not affect the individual in a cumulative way.
- For all facilities, to determine the health impacts, the estimated concentrations were compared against the Emergency Response Planning Guidelines (ERPG) levels 1, 2, and 3 concentration limits or alternates.

F.2.4.2.2.4 Results. The airborne concentrations, averaged over the duration of each exposure, were calculated using EPIcode for the combustion products resulting from the fire for the worker, MCW, NPA, and MOI (as applicable for the accident site) under both 50% and 95% meteorology. The airborne concentrations were compared to respective ERPG values where available. However, ERPG values have not been derived for some of the constituents listed. The effects of these constituents were assessed by comparison with other appropriate values for toxic effects as discussed in Section F.2.3.3.

Tables F.2.4.2-15 through -38 list the downwind concentrations at various locations and corresponding ERPG (or equivalent) values. Results for the diesel fuel fire at fuel examination sites indicate that the toxic chemical concentrations for sulfur dioxide and oxides of nitrogen may exceed ERPG-3 levels for the worker. At Savannah River and Hanford, the MCW also may be exposed to a nitric oxide concentration exceeding ERPG-3 levels under 95% meteorological conditions. The NPA and MOI exposures at all the fuel examination sites would be expected to be below ERPG-2 levels except for Oak Ridge. At this location under 95% meteorological conditions, the NPA and MOI may be exposed to concentrations of sulfur dioxide and oxides of nitrogen that exceed ERPG-3 and concentrations of carbon monoxide that exceed ERPG-2. Under 50% meteorological conditions at Oak Ridge, the NPA and MOI may be exposed to concentrations of nitric oxide that exceed ERPG-3 and concentrations of sulfur dioxide and nitrogen dioxide that exceed ERPG-2. Results for the diesel fuel fire at shipyards show that for the worker and NPA categories, the toxic chemical concentrations for sulfur dioxide and oxides of nitrogen may exceed ERPG-3 levels. For the MOI, however, these concentrations are expected to be less than the ERPG-3 levels with the exception that under 95% meteorological conditions the ERPG-3 level for nitric oxide may be exceeded at the Norfolk shipyard. Results for the ship diesel fuel fire show that shipboard (worker) concentrations of carbon monoxide, sulfur dioxide, and oxides of nitrogen may exceed ERPG-3 levels, but the shore (MOI) concentrations are expected to be less than ERPG-3 levels. For the individuals on board the ship that might be exposed to toxic chemicals at levels above ERPG-3, it is expected that actual toxic chemical exposures would be much less due to the mitigative measures that would be implemented (Section F.2.4.3).

Additional information on the toxic properties for the chemicals that dominate the toxic effects is provided below.

Sulfur dioxide is a colorless gas with a pungent odor. It is a poison, and it is also an eye, skin, and mucous membrane irritant. It chiefly affects the upper respiratory tract and bronchi and at higher concentrations, sulfur dioxide causes respiratory paralysis (TOXnet 1993).

Nitric oxide and **nitrogen dioxide** occur together in dynamic equilibrium. Nitric oxide is a colorless gas, and nitrogen dioxide is a reddish brown gas. Both chemicals are eye, skin, and mucous membrane irritants and primarily affect the respiratory system. Exposure to 47 mg/m^3 of nitrogen dioxide can cause respiratory irritation and chest pain, 93 mg/m^3 can cause lung injuries, and 187 mg/m^3 can be fatal (TOXnet 1993).

In addition to comparing the airborne concentrations to their respective ERPG or other appropriate values, each substance was evaluated to determine if it has the potential for future carcinogenic impacts. It was determined that exposure to lead could result in an increased likelihood for developing cancer. However, sufficient information to quantify this likelihood was not available in IRIS. Therefore, the concentrations of lead resulting from the accident were compared against the NAAQS value for lead. For the lead concentrations provided in Tables F.2.4.2-15 through F.2.4.2-38, no NAAQS is exceeded.

F.2.4.3 Mitigative Measures for Toxic Chemicals. Mitigative measures for potential releases of toxic materials involve administrative controls for personnel protection and emergency response. For personnel protection, controls involve safety review committees for planned activities that establish requirements, safe work permits, and procedures for required clothing (rubber boots, gloves, face shields, eye protection) that can mitigate the effects of potential releases of toxic materials. Procedures may also require provisions for prestationing mitigative devices such as eyewash stations and emergency showers. All of the alternate facilities being evaluated employ emergency response programs to mitigate impacts of potential toxic chemical accidents to workers and the public. Emergency planning, emergency preparedness, and emergency response programs are in place and involve established resources such as warning communications, fire departments, and emergency command centers. The cargo ships used for naval spent nuclear fuel have smoke detection and fire fighting equipment on board. They also have fire suppression systems in their holds which use inert gas to smother fires. In addition, less freely available oxygen in the ship's cargo hold would tend to slow the combustion rate of the diesel fuel. Port facilities would also have available additional fire fighting equipment, public warning systems, and emergency response programs.

Table F.2.4.2-15. Summary of chemical concentrations for fire involving diesel fuel at the INEL Expended Core Facility.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	480	150	2000	220	3.9×10^{-3}
MCW	0.25	7.7×10^{-2}	1.0	0.11	9.5×10^{-7}
MOI	0.15	4.8×10^{-2}	0.65	7.3×10^{-2}	6.1×10^{-7}
NPA	0.16	5.0×10^{-2}	0.69	7.7×10^{-2}	6.1×10^{-7}

Table F.2.4.2-16. Summary of chemical concentrations for fire involving diesel fuel at the INEL Expended Core Facility.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	1200	370	5100	560	4.6×10^{-3}
MCW	1.45	0.45	6.1	0.68	3.0×10^{-7}
MOI	0.66	0.20	2.7	0.30	4.7×10^{-8}
NPA	0.66	0.20	2.7	0.30	4.7×10^{-8}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-17. Summary of chemical concentrations for fire involving diesel fuel at Savannah River.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	200	62	850	94	2.0×10^{-3}
MCW	3.6	1.1	15	1.7	3.6×10^{-5}
MOI	0.13	4.1×10^{-2}	0.55	6.1×10^{-2}	7.5×10^{-7}
NPA	0.13	4.1×10^{-2}	0.55	6.1×10^{-2}	7.5×10^{-7}

Table F.2.4.2-18. Summary of chemical concentrations for fire involving diesel fuel at Savannah River.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	1200	370	5100	560	4.6×10^{-3}
MCW	49	15	200	23	6.9×10^{-5}
MOI	0.90	0.28	3.8	0.42	1.1×10^{-7}
NPA	0.90	0.28	3.8	0.42	1.1×10^{-7}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-19. Summary of chemical concentrations for fire involving diesel fuel at Hanford.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	200	62	840	94	2.0×10^{-3}
MCW	5.2	1.6	21	2.4	4.1×10^{-5}
MOI (New ECF)**	8.3×10^{-2}	2.4×10^{-2}	0.34	3.7×10^{-2}	2.5×10^{-7}
MOI (FMEF)***	0.11	3.3×10^{-2}	0.44	4.9×10^{-2}	8.1×10^{-7}
NPA	0.16	4.8×10^{-2}	0.65	7.3×10^{-2}	7.6×10^{-7}

Table F.2.4.2-20. Summary of chemical concentrations for fire involving diesel fuel at Hanford.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	1200	370	5100	560	4.6×10^{-3}
MCW	32	9.7	130	15	3.9×10^{-5}
MOI (New ECF)**	0.34	0.10	1.4	0.15	4.9×10^{-8}
MOI (FMEF)***	1.0	0.32	4.3	0.48	1.5×10^{-7}
NPA	0.78	0.24	3.2	0.36	5.0×10^{-7}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.
** MOI (New ECF) applies if spent fuel facility is constructed at the 200 Area on the Hanford Site.
*** MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Materials Examination Facility.

Table F.2.4.2-21. Summary of chemical concentrations for fire involving diesel fuel at the Nevada Test Site.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	73	22	300	34	8.3×10^{-4}
MCW	2.3×10^{-2}	7.0×10^{-3}	9.6×10^{-2}	1.1×10^{-2}	2.2×10^{-7}
MOI	8.0×10^{-2}	2.4×10^{-2}	0.33	3.7×10^{-2}	7.3×10^{-7}

Table F.2.4.2-22. Summary of chemical concentrations for fire involving diesel fuel at the Nevada Test Site.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	1200	370	5100	560	4.6×10^{-3}
MCW	1.1	0.34	4.6	0.52	1.7×10^{-7}
MOI	1.4	0.43	5.9	0.65	2.7×10^{-7}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-23. Summary of chemical concentrations for fire involving diesel fuel at Oak Ridge.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	480	150	2000	220	3.9×10^{-3}
MCW	3.8	1.2	16	1.8	3.0×10^{-5}
MOI	37	11	150	18	3.3×10^{-4}
NPA	54	17	230	26	5.0×10^{-4}

Table F.2.4.2-24. Summary of chemical concentrations for fire involving diesel fuel at Oak Ridge.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	1200	370	5100	560	4.6×10^{-3}
MCW	24	7.3	98	11	2.6×10^{-5}
MOI	230	70	950	110	5.3×10^{-4}
NPA	340	100	1400	160	8.7×10^{-4}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-25. Summary of chemical concentrations for fire involving diesel fuel at the Barnwell Plant.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	200	62	840	94	2.0×10^{-3}
MCW	9.5×10^{-2}	2.9×10^{-2}	0.40	4.4×10^{-2}	9.3×10^{-7}
MOI	0.65	0.20	2.7	0.30	5.0×10^{-6}
NPA	0.65	0.20	2.7	0.30	5.0×10^{-6}

Table F.2.4.2-26. Summary of chemical concentrations for fire involving diesel fuel at the Barnwell Plant.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	1200	370	5100	560	4.6×10^{-3}
MCW	2.0	0.62	8.4	0.94	5.4×10^{-7}
MOI	5.8	1.7	24	2.7	3.2×10^{-6}
NPA	5.8	1.7	24	2.7	3.2×10^{-6}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-27. Summary of chemical concentrations for fire involving diesel fuel at Kenneth A. Kesselring Site.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	44	13	180	20	4.8×10^{-4}
MOI	0.25	7.7×10^{-2}	1.0	0.11	2.3×10^{-6}
NPA	0.25	7.7×10^{-2}	1.0	0.11	2.3×10^{-6}

Table F.2.4.2-28. Summary of chemical concentrations for fire involving diesel fuel at Kenneth A. Kesselring Site.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	500	150	2100	230	1.9×10^{-3}
MOI	3.9	1.2	17	1.8	3.1×10^{-6}
NPA	3.9	1.2	17	1.8	3.1×10^{-6}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-29. Summary of chemical concentrations for fire involving diesel fuel at Norfolk Naval Shipyard.

	CHEMICAL CONCENTRATIONS mg/m³ - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	44	13	180	20	4.8×10^{-4}
MOI	4.3	1.3	18	2.0	4.7×10^{-5}
NPA	4.3	1.3	18	2.0	4.7×10^{-5}

Table F.2.4.2-30. Summary of chemical concentrations for fire involving diesel fuel at Norfolk Naval Shipyard.

	CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	500	150	2100	230	1.9×10^{-3}
MOI	47	14	200	22	2.8×10^{-4}
NPA	47	14	200	22	2.8×10^{-4}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-31. Summary of chemical concentrations for fire involving diesel fuel at Pearl Harbor Naval Shipyard.

	CHEMICAL CONCENTRATIONS mg/m³ - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	200	61	830	92	1.6×10^{-3}
MOI	3.3	1.0	13	1.5	1.7×10^{-5}
NPA	12	3.6	49	5.4	1.4×10^{-4}

Table F.2.4.2-32. Summary of chemical concentrations for fire involving diesel fuel at Pearl Harbor Naval Shipyard.

	CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	500	150	2100	230	1.9×10^{-3}
MOI	11	3.4	47	5.3	1.4×10^{-5}
NPA	500	150	2100	230	1.9×10^{-3}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-33. Summary of chemical concentrations for fire involving diesel fuel at Portsmouth Naval Shipyard.

	CHEMICAL CONCENTRATIONS mg/m³ - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	33	10	140	15	3.6×10^{-4}
MOI	1.7	0.51	7.0	0.78	1.7×10^{-5}
NPA	2.7	0.83	11	1.2	3.0×10^{-5}

Table F.2.4.2-34. Summary of chemical concentrations for fire involving diesel fuel at Portsmouth Naval Shipyard.

	CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	500	150	2100	230	1.9×10^{-3}
MOI	24	7.2	99	11	3.7×10^{-5}
NPA	73	22	300	34	1.7×10^{-4}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-35. Summary of chemical concentrations for fire involving diesel fuel at Puget Sound Naval Shipyard.

	CHEMICAL CONCENTRATIONS mg/m³ - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	33	10	140	15	3.6×10^{-4}
MOI	1.5	0.47	6.3	0.71	1.5×10^{-5}
NPA	13	4.0	54	6.1	1.4×10^{-4}

Table F.2.4.2-36. Summary of chemical concentrations for fire involving diesel fuel at Puget Sound Naval Shipyard.

	CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	500	150	2100	230	1.9×10^{-3}
MOI	21	6.5	89	9.8	3.2×10^{-5}
NPA	200	61	830	92	5.8×10^{-4}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

Table F.2.4.2-37. Summary of chemical concentrations for fire involving diesel fuel aboard ship in Puget Sound.

	CHEMICAL CONCENTRATIONS mg/m^3 - 50% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	900	280	3800	420	9.9×10^{-3}
MOI	4.0	1.2	17	1.9	4.1×10^{-5}

Table F.2.4.2-38. Summary of chemical concentrations for fire involving diesel fuel aboard ship in Puget Sound.

	CHEMICAL CONCENTRATIONS mg/m^3 - 95% METEOROLOGY				
	Carbon Monoxide ERPG-1 29 ERPG-2 172 ERPG-3 1720	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Nitric Oxide ERPG-1 31 ERPG-2 * ERPG-3 123	Nitrogen Dioxide ERPG-1 5.6 ERPG-2 9.4 ERPG-3 94	Lead ERPG-1 0.15 ERPG-2 70 ERPG-3 700
Worker	9900	3100	41000	4600	3.8×10^{-2}
MOI	28	8.8	120	13	1.7×10^{-4}

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the ERPG-1 level.

F.3 AIRCRAFT CRASH PROBABILITIES

F.3.1 Introduction

The probability of an airplane crashing into a fuel storage area or a fuel examination facility at the various alternate site locations is presented in this section. An airplane crash into these regions is of concern since it might result in the release of corrosion products from the stored fuel or the release of radioactive fission products from the fuel. The method outlined in "A Methodology for Calculation of the Probability of Crash of an Aircraft into Structures in Weapon Storage Areas" (Sandia 1983) has been used to predict the crash probabilities for this analysis. This calculational methodology takes into consideration the crash probabilities associated with landing and takeoff operations at nearby airports and crashes during in-flight operations.

The aircraft crash probability analysis presented herein is based on the examination of large civilian aircraft and military aircraft crossing the space within a 10-mile radius of each site. The crash probability of general aviation aircraft is not included in this assessment since aircraft of this type generally do not possess sufficient mass or attain sufficiently high velocities to produce a serious radiological threat in the event that they crash into a fuel storage area or a fuel examination facility. Further, the crash probability contribution due to air travel beyond 10 miles was determined to be very small based on the models and conditions used in this analysis, and therefore has been omitted.

F.3.2 Methodology

The Sandia report provides the methodology which has been used for this assessment (Sandia 1983). In this report, the following expressions are given for calculating the crash probability associated with takeoff and landing operations at a given airport runway, and in-flight operations along a given airway:

$$P_{to} = \sum Ni_{to} \bullet Pn_{to} \bullet A \bullet c(a) \bullet e^{-|xij|/\theta(x,a)} \bullet e^{-|yij|/\theta(y,a)}$$

$$P_{l} = \sum Ni_{l} \bullet Pn_{l} \bullet A \bullet c(a) \bullet e^{-|xij|/\theta(x,a)} \bullet e^{-|yij|/\theta(y,a)}$$

$$P_{if} = \sum Nk \bullet Pn_{if} \bullet A \bullet c(if) \bullet e^{-|xkj|/\theta(x,if)}$$

where: subscript "to" refers to airport takeoff operations

subscript "l" refers to airport landing operations

subscript "if" refers to in-flight operations

Ni	=	the number of runway operations per year
Nk	=	the number of in-flight operations per year
Pn	=	the crash probability per operation given in Table F.3-1
x_{ij}	=	the perpendicular distance from the centerline of the runway to the target in miles
x_{kj}	=	the perpendicular distance from the airway to the target in miles
y_{ij}	=	the perpendicular distance from the end of the runway to the target in miles
c(a)	=	crash density constant given in Table F.3-2
c(if)	=	crash density constant given in Table F.3-3
$\theta(x,a)$	=	crash density constant given in Table F.3-2
$\theta(y,a)$	=	crash density constant given in Table F.3-2
$\theta(x,if)$	=	crash density constant given in Table F.3-3
A	=	effective crash area in square miles.

Table F.3-1. Crash parameter Pn.

Operation	Military High Performance	Large Civilian and Military
to	1.6×10^{-6}	0.6×10^{-6}
l	3.1×10^{-6}	2.3×10^{-6}
if	3.9×10^{-8}/mile	0.5×10^{-8}/mile

Table F.3-2. Crash density constants.

Zone (1)	Operation	Military High Performance			Large Civilian and Military		
		c(a)	$\theta(x,a)$	$\theta(y,a)$	c(a)	$\theta(x,a)$	$\theta(y,a)$
I	to	0.043	3.0	3.0	0.28	0.7	1.4
	l	0.11	1.0	3.0	0.28	0.7	1.4
II	to	0	---	---	0	---	---
	l	0.006	1.0	3.0	0.014	0.7	1.4

(1) Refer to Figure F.3-1 for crash zones.

Table F.3-3. Crash density constants.

Operation	Military High Performance		Large Civilian and Military	
	c(if)	$\theta(x,if)$	c(if)	$\theta(x,if)$
if	0.5	1.0	0.8	0.63

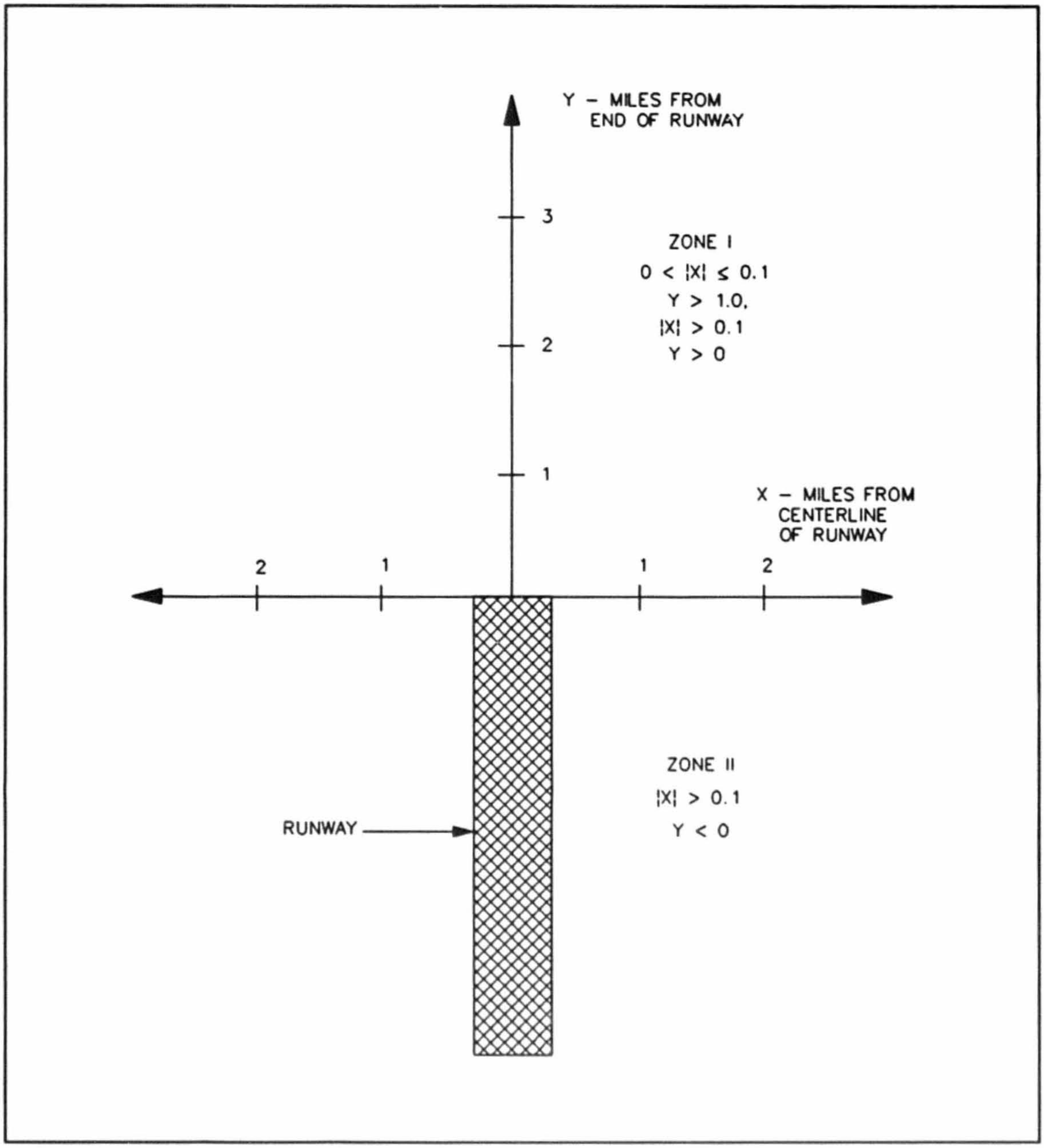

Figure F.3-1. Crash zones.

Using these relationships, the crash probability for takeoff, landing, and in-flight operations is the product of the number of operations per year, times the crash probability per operation per year, times the effective crash area per square mile, times the crash probability density per square statute mile. To determine the crash probability associated with a given site requires the repeated application of these relationships for each airport runway and for each airway. These individual crash components are then summed to arrive at a total overall crash probability for a site.

In the Sandia report, the effective crash area is identified as the sum of the effective skid area of the plane, the effective plan view associated with the target, and the effective shadow area of the crash (Sandia 1983). The following expression relates these terms and is valid for crash attitude angles greater than zero. If the crash attitude angle is zero, an airplane would be flying along parallel to the ground at an altitude equal to or greater than the height of the target; therefore, the airplane would clear the object and there would be no crash.

$$A = (L + A_w) \bullet (W + S_k + H \bullet \cot \phi)$$

where: L = target length dimension

W = target width dimension

H = target height

A_w = aircraft wingspan

ϕ = crash attitude angle

S_k = aircraft skid distance.

F.3.3 Site Specific Information

The existence and location of airports and airways within 10 statute miles of a site have been obtained from Sectional Aeronautical Maps published by the National Oceanic and Atmospheric Administration (NOAA), and from detailed site specific maps which identify nearby airports (NOAA 1993a; NOAA 1993b; NOAA 1993c; NOAA 1993d; NOAA 1993e; NOAA 1993f; NOAA 1993g; USGS 1983a; USGS 1983b). These same sources of information were also used to obtain the distances from airport runways and airways to the sites of interest. Information regarding

air traffic along airways within this region was obtained from the Federal Aviation Administration (FAA). Airplane holding patterns and approach and departure routes that were identified by the FAA were converted into equivalent airways for this analysis. Information regarding the number of takeoff and landing operations at each airport runway was obtained from the cognizant airport officials (i.e., airport manager or base commander), or from the FAA. Tables F.3-4 and F.3-5 summarize the airport and airway traffic information that was obtained.

Table F.3-4. Airport landings and takeoffs per site location per year.

Site Location	Airport	Large Civilian Aircraft		Large Military Aircraft		Military High Performance Aircraft	
		No. Landings	No. Takeoffs	No. Landings	No. Takeoffs	No. Landings	No. Takeoffs
Barnwell Plant	Barnwell County	0	0	0	0	0	0
Hanford 200 Area	None	-	-	-	-	-	-
400 Area	Richland County	0	0	0	0	0	0
INEL	(1)	150	150	0	0	0	0
Kesselring	Saratoga County	0	0	0	0	0	0
Nevada Test Site	None	-	-	-	-	-	-
Norfolk	Norfolk Intl	21200	21200	0	0	0	0
	Chambers	850	850	6600	6600	11100	11100
Oak Ridge	None	-	-	-	-	-	-
Pearl Harbor	Honolulu Intl / Hickam Air Force Base	101300	101300	5750	5750	8650	8650
	Barbers Point NAS	0	0	20500	20500	850	850
	Ford Island	0	0	0	0	0	0
Portsmouth	Pease Intl	16400(3)	16400(3)	2450(2)	2450(2)	2450(2)	2450(2)
	Little-brook	0	0	0	0	0	0
Puget Sound	Bremerton Natl	4(4)	4(4)	4(4)	4(4)	4(4)	4(4)
	Apex	0	0	0	0	0	0
	Port Orchard	0	0	0	0	0	0
Savannah River	None	-	-	-	-	-	-

(1) FAA testing of new commercial aircraft at NOAA tower.
(2) Split between aircraft types is estimated to be equal. Precise breakdown not furnished by airport.
(3) Operations based on total civilian aircraft. Breakdown of only large aircraft not furnished by airport.
(4) Operations based on this aircraft type being available only during annual air show.

Table F.3-5. Airway air traffic per site location per year.

Site Location	Large Civilian	Large Military	Military High Performance
Barnwell Plant	5900	2600	3300
Hanford 200 Area	2200	0	0
400 Area	3200	100	0
INEL	0	0	0
Kesselring	98600	144	0
Nevada Test Site	22000	9000	19000
Norfolk	17000	350	550
Oak Ridge	86900	5900	4700
Pearl Harbor	0	0	1750
Portsmouth	11000	0	0
Puget Sound	12800	0	0
Savannah River	5900	2600	3300

The effective crash area associated with various types of fuel storage at shipyards and prototypes was based on the storage facility footprints identified in Table D-1 of Attachment D. Length and width dimensions associated with the target area were calculated from these footprints by treating the storage area as square (i.e., equal length and width dimensions). The height of the dry storage containers was based on that of an existing M-140 shipping container, and the height of the water pool facility superstructure was based on the approximate height of the Expended Core Facility at INEL. For the water pool facility, a crash into the building might damage the fuel either by the airplane directly striking it or by the airplane causing sufficient damage to the building to cause part of the building structure to collapse and strike the fuel. The crash attitude angle used was 15 degrees, based on the recommended value identified in the Sandia report (Sandia 1983). A reduced aircraft skid distance of 300 feet was used. This skid distance is based on a review of the proposed site locations and reflects the fact that nearby buildings, dry docks, or retaining walls will generally limit the length of the aircraft skid to 300 feet or less prior to impact.

The effective crash area associated with fuel examination at the Expended Core Facility at INEL or similar facilities to be constructed at the Barnwell Plant, Hanford, Oak Ridge, the Nevada Test Site, or Savannah River was based on the vulnerable part of the facility being 667 feet long, 194 feet wide, and 60 feet high. This represents the portion of the Expended Core Facility that contains

the combined dry cell, shielded cell, and water pool as identified in Attachment B. For these facilities, a crash into the building might damage the fuel either by the airplane directly striking it or by the airplane causing sufficient damage to the building to cause part of the building structure to collapse and strike the fuel. The effective crash area associated with dry storage or shipping containers waiting to be handled at these fuel examination facilities is based on the height and width of an existing M-140 shipping container and the modeling approach that two such containers could be located outside of the fuel processing facility and separated by a reasonably large distance. The crash attitude angle that was used was 15 degrees. For these facilities and containers, airplane skid distances of 2200 feet for military high performance aircraft and 1600 feet for large military and large civilian aircraft were used. These skid distances correspond to the maximum expected skid distance based on the information presented in the Sandia report (Sandia 1983).

F.3.4 Aircraft Specific Information

Aircraft wingspans which are representative of large civilian aircraft, military high performance aircraft (i.e., tactical fighter and tactical fighter trainer), and large military aircraft (i.e., cargo, transport, refueling, and bomber) have been taken into account separately in computing the overall crash probabilities for each site. Wingspans for these three class of aircraft have been based on average values computed from individual planes within each class. Data from "Aviation Week & Space Technology" served as the basis for determining these wingspans (AWST 1992). The calculated average wingspans were: 40 feet for military high-performance aircraft, 131 feet for large military aircraft, and 135 feet for large civilian aircraft. For large military and civilian aircraft, an effective wingspan that was 75% of the average wingspan was used in the probability calculations. This effective wingspan reflects the fact that only the region between the most outboard wing-mounted engines has the potential to seriously damage a fuel storage area or a fuel examination facility.

F.3.5 Results

Tables F.3-6 and F.3-7 present the crash probability results for the four methods of fuel storage at shipyards and prototypes and for fuel examination facilities. The probabilities listed within these tables represent the combined takeoff, landing, and in-flight crash probabilities associated with each method of fuel storage at each site. Following the DOE NEPA oversight guidance, consequences for beyond design basis accidents are calculated where the probability is 10^{-7} or greater per year. These consequences are discussed in Section F.1.4 of this attachment. For cases less likely than 10^{-7} per year, calculations of consequences are not included.

The probability calculated for airplane crashes at different facilities located within a particular DOE site may vary somewhat. This situation exists at INEL where low altitude testing of commercial jet airliners has been conducted near the NOAA tower. This tower is located about 1.5 miles from ICPP, and 2.3 miles from ECF. As a result of this difference in distance, the crash probabilities are expected to be about a factor of two higher at ICPP than at ECF. Further, two different methodologies have been in general use for determination of aircraft accident probabilities. In addition to the Sandia methodology used in this appendix, a technique developed by the NRC in the 1970's has been applied at some facilities. Comparison of the two methods has shown that results can differ by a factor of two to four, with the NRC method generally producing higher probabilities than the Sandia method. This difference stems from the somewhat more detailed nature of the Sandia method. Therefore, calculated aircraft crash probabilities at ICPP are expected to be about a factor of four to eight higher than those calculated for ECF.

Crash probabilities fall in the design basis range (i.e., probability of occurrence $\geq 10^{-6}$ per year) at Pearl Harbor for all types of fuel storage, at Norfolk for fuel storage in shipping containers on railcars, and at Oak Ridge and Savannah River for the fuel examination facility dry cell and water pool. The radiological consequences associated with an airplane crash into these areas are addressed in detail in Section F.1.4.

Crash probabilities fall in the beyond design basis range (i.e., probability of occurrence between 10^{-6} and 10^{-7} per year) at Norfolk for fuel storage in immobile dry storage containers, shipping containers on a concrete pad, and in the water pool facility, at Kesselring for fuel storage in shipping containers on railcars and in the water pool facility, at Portsmouth for shipping containers on railcars, at the Nevada Test Site for the fuel examination facility dry cell and water pool, and the fuel examination facility dry storage containers at Oak Ridge and Savannah River. The radiological consequences associated with an airplane crash into these areas are also addressed in detail in Section F.1.4.

Crash probabilities with a likelihood of occurrence less than 10^{-7} per year are not evaluated since it is expected that they would contribute very very little to the risk. This is the case for immobile dry storage and shipping containers on a concrete pad at Kesselring and Portsmouth, the

water pool facility at Portsmouth, all types of fuel storage at Puget Sound, the fuel examination facilities at Barnwell, Hanford, and INEL, and the fuel examination facility dry storage containers at the Nevada Test Site.

Table F.3-6. Crash probabilities for various fuel storage options per site location per year.

Site Location	Immobile Dry Storage Containers	Shipping Containers on Concrete Pad	Shipping Containers on Railcars	Water Pool Facility
Kesselring	9×10^{-8}	8×10^{-8}	1×10^{-7}	2×10^{-7}
Norfolk	6×10^{-7}	[illegible] $\times 10^{-7}$	1×10^{-6}	4×10^{-7}
Pearl Harbor	1×10^{-5}	1×10^{-5}	N/A	2×10^{-5}
Portsmouth	6×10^{-8}	6×10^{-8}	1×10^{-7}	7×10^{-8}
Puget Sound	3×10^{-8}	3×10^{-8}	8×10^{-8}	3×10^{-8}

Table F.3-7. Crash probabilities for fuel examination facilities per site location per year.

Site Location	Shielded Cell, Dry Cell, and Water Pool	Dry Storage Containers
Barnwell Plant	9×10^{-8}	1×10^{-8}
Hanford 200 Area	6×10^{-10}	2×10^{-10}
400 Area	4×10^{-8}	1×10^{-8}
INEL (ECF)	7×10^{-8}	2×10^{-8} 5×10^{-8} (1)
Nevada Test Site	4×10^{-7}	5×10^{-8}
Oak Ridge	1×10^{-6}	3×10^{-7}
Savannah River	2×10^{-6}	3×10^{-7}

(1) Crash probability based on 582 dry storage containers stored in a square array several hundred yards away from ECF. Array footprint is 168,800 square feet.

F.4 FUGITIVE DUST

The INEL-ECF is a large laboratory facility used to receive, examine, and ship naval nuclear fuel and irradiated test specimen assemblies. This section provides the results of an evaluation of fugitive dust emissions that could be generated during the construction of a similar laboratory facility at an alternate location (Hanford, Savannah River, the Nevada Test Site, the Barnwell Plant, or Oak Ridge).

F.4.1 Computer Modeling to Estimate Fugitive Dust Emissions

Factors such as locations of affected persons, terrain, meteorological conditions, release conditions, and grain size distributions are required as input parameters for calculations to determine particulate concentrations from fugitive dust emissions during construction activities. This section describes the computer model used to perform fugitive dust concentration estimates. Specific input parameters used in this analysis are summarized in Section F.4.2.

The Fugitive Dust Model (FDM) was the computer code chosen to evaluate fugitive dust emissions from construction activities at an alternate DOE location. FDM is a computerized air quality model specifically designed for estimating fugitive dust emissions from point, line, or area sources (EPA 1992c).

FDM is designed to work with properly prepared meteorological data such as the EPA RAMMET program or card images of meteorological data in either hourly or Stability Array (STAR) format. FDM is based on the well-known Gaussian plume formulation for computing concentrations, but the model has been specifically adapted to incorporate an improved gradient transfer deposition algorithm. Emissions for each source are apportioned by the user into a series of particle size classes. A gravitational settling velocity and a deposition velocity are subsequently calculated by FDM for each class, and dust concentrations and depositions are then calculated for locations selected by the user.

FDM is the preferred model for estimating conditions resulting from particulate matter emissions from fugitive sources such as excavation and soil handling. The ISC2 Code (Section

F.2.2.2) can also be used for this purpose; however, FDM was judged to be superior to the ISC2 Code for this evaluation.

F.4.2 Conditions and Key Parameters

- Construction area was 30 acres.
- Construction activities occurred over a 3- to 5-year period.
- An emission factor of 2.0 tons per acre-month was used.
- Grain sizes used were as follows:

Average Diameter (μm)	% of Total
1.25	3
3.75	5
7.5	15
12.5	10
20.0	67

- Meteorological conditions used were the 5-year average STAR data sets.
- Roughness heights were 2 centimeters for Hanford and Nevada Test Site and 30 centimeters for Savannah River, the Barnwell Plant, and Oak Ridge.

F.4.3 Results

The fugitive dust concentrations were calculated using FDM for the worker, MCW, NPA, and MOI using normal meteorology. Table F.4-1 lists the fugitive dust concentrations at various locations. These airborne concentrations were compared against the TLV-TWA concentration for particulates. The TLV-TWA concentration of 10 mg/m^3 was not exceeded at any of the specified locations for fugitive dust that could be generated during construction activities at the alternate locations. Since these concentrations were extremely low, it can also be concluded that similar results would be expected for the alternate shipyard locations since the facilities to be constructed would be smaller.

Table F.4-1. Summary of fugitive dust concentrations for construction activities at alternate locations.

Fugitive Dust Concentration mg/m^3					
	Savannah River	Hanford*	Nevada Test Site	Oak Ridge	Barnwell Plant
Worker	2.7	3.5	1.6	3.1	2.7
MCW	3.6×10^{-2}	7.3×10^{-2}	8.1×10^{-5}	2.9×10^{-3}	5.2×10^{-4}
MOI	2.8×10^{-4}	1.3×10^{-4}	2.9×10^{-4}	0.22	3.2×10^{-3}
NPA	1.4×10^{-4}	2.2×10^{-4}	Not applicable	1.6	3.2×10^{-3}

*MOI shown is for a new spent fuel facility constructed at the 200 Area on the Hanford Site. The MOI concentration is 3.0×10^{-4} mg/m^3 for a new spent fuel facility constructed at the Fuels and Materials Examination Facility.

F.5 OCCUPATIONAL ACCIDENTS

Occupational accidents can occur in the workplace during the construction or operation of any industrial facility. In order to assess the possible extent of occupational accidents during construction and non-construction operations at naval spent nuclear fuel facilities, projections of the number of fatalities and injuries or illnesses were made for each alternative. The projections are presented in this section. The projections are based on average occupational fatality and injury incidence rate data published by the DOE (DOE 1993a) for DOE and DOE contractor operations. The incidence rates that were used in the analyses are provided below. A more detailed discussion of the basis for these incidence rates is presented in Volume 1.

Average occupational injury/illness and fatality rates[a]

	All Labor Categories		Construction Workers	
	Total Injury/Illness	Fatalities	Total Injury/Illness	Fatalities
DOE and Contractors[b]	3.2	0.0032	6.2	0.011

[a] All incidence rates are given per 100 worker-years
[b] 1988-1992 averages (DOE 1993a)

The term "injury/illness" as used in this analysis corresponds to the DOE definition of a recordable injury illness. Specifically, an injury or illness case represents any work-related death, illness, or any work-related injury which would result in loss of consciousness, restriction of work or motion, transfer to another job, or medical treatment beyond first aid.

F.5.1 Accident Evaluation

F.5.1.1 Construction. The average number of construction-related fatalities and injury or illnesses and the 40-year total were calculated. The methods of calculating construction-related fatalities and injuries or illnesses are presented below.

The number of construction workers that would be required to construct or modify each naval spent nuclear fuel storage and examination facility was calculated for every year that construction would take place during the period 1995 through 2035. The sum of these workers represents the total number of construction workers. The 40-year total of construction fatalities was obtained by multiplying the total number of construction workers by the construction fatality rate for DOE and DOE contractors.

The annual average number of construction workers for each facility was obtained by dividing the total number of construction workers by the number of years that construction would take place. The product of the annual average number of construction workers and the construction fatality rate for DOE and DOE contractors was calculated to provide the annual average number of construction fatalities.

The annual average and 40-year total construction injuries or illnesses were calculated in the same manner as construction fatalities except that the construction injury or illness accident rate for DOE and DOE contractors.

F.5.1.2 Storage and Examination Facility Operations. The average number of fatalities and injuries or illnesses and the 40-year total fatalities and injuries or illnesses were calculated for operation of naval spent nuclear fuel storage and examination facilities. The methods of calculating the operational fatalities and injuries or illness are presented below.

The accident rates for DOE and DOE contractor operations other than construction were used because examination and storage facility operations would more likely be performed by DOE and DOE contractor personnel (or Navy personnel in the case of shipyards). The number of workers that would be required to operate each naval spent nuclear fuel storage and examination facility was calculated for every year during the period 1995 through 2035 and summed over the 40-year period to obtain the total number of workers. The 40-year total of fatalities was obtained by multiplying the total number of workers by the DOE fatality rate.

The annual average number of workers for each facility was obtained by dividing the total number of workers by the number of operational years (40 years). The product of the annual average number of workers and the DOE fatality rate represents the annual average number of operational fatalities.

The annual average and 40-year total estimated injuries or illnesses associated with facility operations were calculated in the same manner as fatalities associated with facility operations except that the DOE injury or illness accident rate was used.

F.5.2 Results

This section presents tabulated results of calculations of construction and operating fatalities and injuries or illnesses for each alternative. Table F.5-1 provides the projections of occupational fatalities and injuries or illnesses for construction activities and storage and examination operations for each alternative. Tables F.5-2 through F.5-5 present the results of calculations of occupational fatalities and injuries or illnesses for construction activities and storage and examination operations at naval sites. The results of all calculations show that the number of fatalities and injuries or illnesses for construction activities and storage and examination operations would be low for any alternative.

Table F.5-1. Occupational fatalities and injuries/illnesses by alternative - construction activities and storage and examination facility operations.

	Fatalities				Injuries/Illnesses			
Alternative	Construction		Operations		Construction		Operations	
	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total
1. No Action	3.9×10^{-3}	6.9×10^{-3}	2.5×10^{-3}	9.8×10^{-2}	2.2	3.9	2.5	9.8×10^{1}
2. Decentralization[1]								
• No Exam	3.1×10^{-2}	2.2×10^{-1}	6.6×10^{-3}	2.6×10^{-1}	1.8×10^{1}	1.2×10^{2}	6.6	2.6×10^{2}
• Limited Exam	4.2×10^{-2}	2.5×10^{-1}	8.3×10^{-3}	3.3×10^{-1}	2.4×10^{1}	1.4×10^{2}	8.3	3.3×10^{2}
• Full Exam	3.4×10^{-2}	2.2×10^{-1}	2.1×10^{-2}	8.3×10^{-1}	1.9×10^{1}	1.3×10^{2}	2.1×10^{1}	8.3×10^{2}
3. 1992/1993 Planning Basis	2.6×10^{-3}	5.3×10^{-3}	1.7×10^{-2}	6.6×10^{-1}	1.5	3.0	1.7×10^{1}	6.6×10^{2}
4. Regionalization								
• INEL	2.6×10^{-3}	5.3×10^{-3}	1.7×10^{-2}	6.6×10^{-1}	1.5	3.0	1.7×10^{1}	6.6×10^{2}
• Nevada Test Site	4.7×10^{-2}	3.3×10^{-1}	1.7×10^{-2}	6.7×10^{-1}	2.7×10^{1}	1.9×10^{2}	1.7×10^{1}	6.7×10^{2}
• Oak Ridge	4.7×10^{-2}	3.3×10^{-1}	1.7×10^{-2}	6.7×10^{-1}	2.7×10^{1}	1.9×10^{2}	1.7×10^{1}	6.7×10^{2}
5. Centralization								
• INEL	2.6×10^{-3}	5.3×10^{-3}	1.7×10^{-2}	6.6×10^{-1}	1.5	3.0	1.7×10^{1}	6.6×10^{2}
• Hanford	4.7×10^{-2}	3.3×10^{-1}	1.7×10^{-2}	6.7×10^{-1}	2.7×10^{1}	1.9×10^{2}	1.7×10^{1}	6.7×10^{2}
• Savannah River	4.7×10^{-2}	3.3×10^{-1}	1.7×10^{-2}	6.7×10^{-1}	2.7×10^{1}	1.9×10^{2}	1.7×10^{1}	6.7×10^{2}
• Nevada Test Site	4.7×10^{-2}	3.3×10^{-1}	1.7×10^{-2}	6.7×10^{-1}	2.7×10^{1}	1.9×10^{2}	1.7×10^{1}	6.7×10^{2}
• Oak Ridge	4.7×10^{-2}	3.3×10^{-1}	1.7×10^{-2}	6.7×10^{-1}	2.7×10^{1}	1.9×10^{2}	1.7×10^{1}	6.7×10^{2}

[1] The water pool storage mode was used in the calculation since the maximum number of construction and operational workers would be involved.

440

Table F.5-2. Occupational fatalities for construction activities at Naval Nuclear Propulsion Program sites.

	ECF		Puget Sound		Pearl Harbor		Portsmouth		Norfolk		Kesselring	
Storage	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total
Storage Modes												
1. Railcar Storage			7.7×10^{-4}	2.3×10^{-3}	7.7×10^{-4}	7.7×10^{-4}	7.7×10^{-4}	7.7×10^{-4}	7.7×10^{-4}	2.3×10^{-3}	7.7×10^{-4}	7.7×10^{-4}
2. Shipping Containers on Concrete Pads			3.3×10^{-4}	6.6×10^{-4}	3.3×10^{-4}	6.6×10^{-4}	3.3×10^{-4}	6.6×10^{-4}	3.3×10^{-4}	6.6×10^{-4}	3.3×10^{-4}	6.6×10^{-4}
3. Immobile Storage Containers			3.3×10^{-4}	6.6×10^{-4}	3.3×10^{-4}	6.6×10^{-4}	3.3×10^{-4}	6.6×10^{-4}	3.3×10^{-4}	6.6×10^{-4}	3.3×10^{-4}	6.6×10^{-4}
4. Water Pool Storage			8.1×10^{-3}	5.7×10^{-2}	5.3×10^{-3}	3.7×10^{-2}	5.3×10^{-3}	3.7×10^{-2}	7.7×10^{-3}	5.4×10^{-2}	4.9×10^{-3}	3.4×10^{-2}
Examination Modes												
1. Full Exam	2.6×10^{-3}	5.3×10^{-3}										
2. Limited Exam			1.1×10^{-2}	3.2×10^{-2}								

441

Table F.5-3. Occupational fatalities for storage and examination facility operations at Naval Nuclear Propulsion Program sites.

	ECF		Puget Sound		Pearl Harbor		Portsmouth		Norfolk		Kesselring	
Storage	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total
Storage Modes[1]												
1. Railcar Storage			1.9×10^{-5}	7.7×10^{-4}	1.9×10^{-5}	7.7×10^{-4}	1.9×10^{-5}	7.7×10^{-3}	1.9×10^{-5}	7.7×10^{-4}	1.9×10^{-5}	7.7×10^{-4}
2. Shipping Containers on Concrete Pads			2.7×10^{-5}	1.1×10^{-3}	2.0×10^{-5}	8.0×10^{-4}	2.0×10^{-5}	8.0×10^{-4}	3.3×10^{-5}	1.3×10^{-3}	1.9×10^{-5}	7.7×10^{-4}
3. Immobile Storage Containers			2.2×10^{-4}	8.8×10^{-3}	1.2×10^{-4}	4.8×10^{-3}	1.3×10^{-4}	5.0×10^{-3}	2.6×10^{-4}	1.0×10^{-2}	6.9×10^{-5}	2.8×10^{-3}
4. Water Pool Storage			1.0×10^{-3}	4.1×10^{-2}	8.0×10^{-4}	3.2×10^{-2}	8.1×10^{-4}	3.2×10^{-2}	9.5×10^{-4}	3.8×10^{-2}	6.3×10^{-4}	2.5×10^{-2}
Examination Modes												
1. Full Exam	1.7×10^{-2}	6.6×10^{-1}										
2. Limited Exam			1.8×10^{-3}	7.1×10^{-2}								

[1] Decentralization (No Exam) used for representative case.

442

Table F.5-4. Occupational injuries/illnesses for construction activities at Naval Nuclear Propulsion Program sites.

	ECF		Puget Sound		Pearl Harbor		Portsmouth		Norfolk		Kesselring	
Storage	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total
<u>Storage Modes</u>												
1. Railcar Storage			4.3×10^{-1}	1.3×10^{0}	4.3×10^{-1}	4.3×10^{-1}	4.3×10^{-1}	4.3×10^{-1}	4.3×10^{-1}	1.3×10^{0}	4.3×10^{-1}	4.3×10^{-1}
2. Shipping Containers on Concrete Pads			1.9×10^{-1}	3.7×10^{-1}	1.9×10^{-1}	3.7×10^{-1}	1.9×10^{-1}	3.7×10^{-1}	1.9×10^{-1}	3.7×10^{-1}	1.9×10^{-1}	3.7×10^{-1}
3. Immobile Storage Containers			1.9×10^{-1}	3.7×10^{-1}	1.9×10^{-1}	3.7×10^{-1}	1.9×10^{-1}	3.7×10^{-1}	1.9×10^{-1}	3.7×10^{-1}	1.9×10^{-1}	3.7×10^{-1}
4. Water Pool Storage			4.6×10^{0}	3.2×10^{1}	3.0×10^{0}	2.1×10^{1}	3.0×10^{0}	2.1×10^{1}	4.4×10^{0}	3.1×10^{1}	2.8×10^{0}	1.9×10^{1}
<u>Examination Modes</u>												
1. Full Exam	1.5×10^{0}	3.0×10^{0}										
2. Limited Exam			5.9×10^{0}	1.8×10^{1}								

443

Table F.5-5. Occupational injuries/illnesses for storage and examination faci.ity operations at Naval Nuclear Propulsion Program sites.

	ECF		Puget Sound		Pearl Harbor		Portsmouth		Norfolk		Kesselring	
Storage	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total	Annual Average	40-Year Total
<u>Storage Modes</u>[1]												
1. Railcar Storage			1.9×10^{-2}	7.7×10^{-1}	1.9×10^{-2}	7.7×10^{-1}	1.9×10^{-2}	7.7×10^{-1}	1.9×10^{-2}	7.7×10^{-1}	1.9×10^{-2}	7.7×10^{-1}
2. Shipping Containers on Concrete Pads			2.7×10^{-2}	1.1	2.0×10^{-2}	8.0×10^{-1}	2.0×10^{-2}	8.0×10^{-1}	3.3×10^{-2}	1.3	1.9×10^{-2}	7.7×10^{-1}
3. Immobile Storage Containers			2.2×10^{-1}	8.8	1.2×10^{-1}	4.8	1.3×10^{-1}	5.0	2.6×10^{-1}	1.0×10^{1}	6.9×10^{-2}	2.8
4. Water Pool Storage			1.0	4.1×10^{1}	8.0×10^{-1}	3.2×10^{1}	8.1×10^{-1}	3.2×10^{1}	9.5×10^{-1}	3.8×10^{1}	6.3×10^{-1}	2.5×10^{1}
<u>Examination Modes</u>												
1. Full Exam	1.7×10^{1}	6.6×10^{2}										
2. Limited Exam			1.8	7.1×10^{1}								

(1) Decentralization (No Exam) used for representative case.

444

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ATTACHMENT G

COMPARISON OF THE NAVAL SPENT NUCLEAR FUEL STORAGE ENVIRONMENTAL ASSESSMENT AND THIS ENVIRONMENTAL IMPACT STATEMENT

The Naval Nuclear Propulsion Program has prepared an environmental assessment of short-term storage of naval spent nuclear fuel until the environmental impact statement, of which this appendix is a part, can be completed and an alternative for management of naval spent nuclear fuel is selected (Federal Register, Vol. 59, No. 19, 4051, January 8, 1994). The environmental assessment considered alternatives for storing, until June 1995, naval spent nuclear fuel removed from nuclear-powered vessels and reactor prototypes at several naval sites. The environmental impact statement, which the appendix including this attachment is a part, considers alternatives for the examination and storage of naval spent nuclear fuel during a 40-year period beginning in June 1995.

Occasions may arise when comparison of the impacts for naval spent nuclear fuel described in these two documents may be desired. However, there are some differences between the environmental assessment and this appendix which should be recognized because they make such a comparison complicated. Failure to recognize these differences may lead to an erroneous conclusion that the two documents are inconsistent or contradictory.

First, and most importantly, the environmental assessment considered only a limited period, less than 2 years, needed to conduct the National Environmental Policy Act process required to reach a decision on the long-term management of Department of Energy (DOE) spent nuclear fuel. This process includes preparation of this environmental impact statement. The environmental impact statement, and therefore this appendix, provides the evaluation of the alternatives to be used for managing spent nuclear fuel for 40 years. As a result, this environmental impact statement considers a wider range of alternatives than the environmental assessment, partly because more alternatives are possible if a longer time is available to implement them and partly because some decisions which could be deferred for a short period such as 2 years should not be deferred for a period as long as 40 years.

The alternatives considered in the environmental impact statement also include more potential sites for management of naval spent nuclear fuel. This provides a wider range of choices, but, as a natural consequence, it also increases the number of potential destinations and the miles traveled by shipments of naval spent nuclear fuel under some alternatives. In the same manner, while the environmental assessment considered temporary storage of naval spent nuclear fuel at Newport News Shipbuilding, storage at Newport News is not included in the alternatives in the environmental impact statement because that shipyard is not federally owned.

The alternatives considered in the environmental impact statement also include storage of naval spent nuclear fuel in water pools and immobile dry storage casks in addition to storage in shipping containers. There is also an evaluation of alternatives for examination of naval spent nuclear fuel in the environmental impact statement. These additional storage modes and examination alternatives were not considered in detail in the environmental assessment because the period covered by that document was short and consequently, the implementation of some of the alternatives would have been impractical. For example, water pool storage facilities could not be funded and constructed at the shipyards in a period of less than 2 years.

Also, as a natural result of the longer period considered in this environmental impact statement, a larger number of naval spent nuclear fuel assemblies and additional types of naval fuel assemblies are included in the analyses. The increase in the amount of naval spent nuclear fuel occurs since a certain number of naval reactors are refueled or defueled each year, so in a greater number of years more fuel becomes available for storage. Similarly, some newer designs for naval nuclear propulsion plants will not be refueled for the first time until some time after 1995, so those types of fuel are not treated in the environmental assessment.

The environmental impact statement addresses some impacts of normal operations and some accidents not discussed in the environmental assessment because the conditions or operation which might cause these effects would not occur under the alternatives considered in the environmental assessment. The environmental impact statement also addresses several types of impacts for each alternative in greater detail than the environmental assessment. This was done because more detailed treatment was judged to be appropriate with the broader scope of alternatives in the environmental impact statement.

The methods used to perform the analyses in the environmental impact statement have been refined in the time since the environmental assessment was prepared. This occurred partly because of the larger number of naval spent nuclear fuel assemblies analyzed and the wider scope of sites and methods of storage to be evaluated, and partly because additional time was available to implement the refinements. In addition to refinements in the methods for performing the calculations, some minor changes in the calculational models were made in order to establish a high degree of consistency with the analytical methods used for the other DOE sites that are part of the environmental impact statement. This consistency is appropriate in some cases in order to establish common grounds for comparison of alternatives. The changes in the calculational methods make a direct comparison of the analytical results presented in the environmental assessment for naval sites with those in this appendix difficult.

GLOSSARY

activation	The process of making a material radioactive by exposing the material to neutrons, protons, or other nuclear particles.
activation products	The radionuclides formed as a result of a material being activated. For example, cobalt-60 is an activation product resulting from neutron activation of cobalt-59.
activity	A measure of the rate at which a material is emitting nuclear radiation. Activity is usually measured in terms of the number of nuclear disintegrations which occur in a quantity of the material over a period of time. The standard unit of activity is the curie (Ci), which is equal to 37 billion (3.7×10^{10}) disintegrations per second.
aggregates	Sand, gravel, or rock which is used in concrete or mortar mixes to achieve increased strength.
airborne emissions	Radioactivity in the form of radioactive particles, gases, or both that is transported by air.
alloy	A mixture of two or more metals.
aquifer	A water-bearing stratum of permeable rock, sand, or gravel located beneath the surface of the earth, which is capable of yielding water to a well or spring.
archaeological areas	Areas of or relating to the scientific study of material remains (as fossil relics, artifacts, monuments) of past human life and activities.
average individual	An individual who could consume items or occupy areas at rates which would be typical for the population of interest.
base flood	A flood which has a 1-percent chance of occurrence in any given year. Also referred to as a 100-year flood.
benthic	Pertaining to the bottom of the ocean.
best estimate	An estimate in which the factors used in determining the estimate were chosen such that the result approximately represents what would be expected.
cladding	A metal casing that surrounds the nuclear fuel.

coastal zone	The region along the shore, adjacent to the ocean. A coastal zone is usually defined as the region within 3 nautical miles of a shoreline.
concentration factor	A factor which is defined as the concentration of an element or radionuclide in an organism or its tissues divided by the concentration directly available from the organism's environment under equilibrium or steady-state conditions.
conservative estimate	An estimate in which the factors used in determining the estimate were chosen such that the result would be unlikely to be exceeded.
containments	Devices as complex as a glove box or as simple as a plastic bag designed to limit the spread of radioactive contamination to an area as close as possible to the source, and to break the chain of transfer to prevent contaminating other material.
core	The central portion of a nuclear reactor containing the nuclear fuel.
corrosion	The process denoting the destruction of metal by chemical or electrochemical action.
corrosion products	The substances produced by corrosion of a metal. Rust is a common corrosion product resulting from the corrosion of iron.
corrosion-resistant alloy	An alloy which corrodes slowly compared to ordinary alloys. Stainless steel is an example of a corrosion-resistant alloy.
critical organ	The limiting organ for evaluating exposure to ionizing radiation. A critical organ is determined by the following criteria: (1) the organ that accumulates the greatest concentration of a radioactive material, (2) the necessity of the organ to the well being of the entire body, (3) the organ most damaged by the entry of a radionuclide into the body, and (4) the organ damaged by the lowest exposure. Usually, case (1) is the determining factor for choosing the critical organ.
critical pathways	Those pathways which result in the most significant amount of exposure to radiation.
cumulative effects	The changes in the health of an individual(s) from the sum of all yearly exposures to radiation.

curie (Ci)	The curie is the common unit used for expressing the magnitude of radioactive decay in a sample containing radioactive material. Specifically, the curie is that amount of radioactivity equal to 3.7×10^{10} (37 billion) disintegrations per second. This unit does not give any indication of the radiological hazard associated with the disintegration.
defueling	Removal of all nuclear fuel from a nuclear-powered ship.
design earthquake	The maximum intensity earthquake that might occur along the nearest fault to a structure. Structures are built to withstand a design earthquake.
diffusion	The process of spreading out or scattering from regions of higher concentration to regions of lower concentration.
dispersion	The process of scattering or distributing over a large region.
dose	A general term which denotes the quality of radiation or energy absorbed; usually expressed in rems for doses to man.
dose commitment	The total radiation dose accrued by an individual over a specified period of time due to the exposure of the individual to radiation during a given interval of time. This includes the total time the radioactive material would reside in the body, if ingested or inhaled (usually expressed in rems).
dose commitment conversion factor	A factor which converts the quantity of radioactivity taken into the body to the dose to the individual (usually expressed in rems per curie).
dose equivalent	A quantity used to express all radiations on a common scale for calculating the effective absorbed dose. It is defined as the product of the absorbed dose and certain modifying factors and is expressed in rems.
dose rate	The amount of radiation dose delivered in a unit amount of time; for example, in rems per hour.
dose rate conversion factor	A factor which converts the exposure to a given radiation level to the dose that an individual could receive. It is usually expressed in rems per hour per curie per cubic meter (or square meter).

dredge spoil	Bottom sediments or materials that have been excavated from a waterway.
ecosystem	A community of plant and animal populations together with their physical environment. An organizational unit which can maintain its biological activities independent of other units.
element	A chemical substance that cannot be divided into simpler substances by chemical means. A substance whose atoms all have the same atomic number.
endangered species	A species or subspecies which is in danger of extinction throughout all or a significant portion of its range.
environmental consequences	Changes to the environment as a result of the effects of radiation or radioactive materials.
epidemiological study	A scientific study that deals with the incidence, distribution, and control of disease in a specified population.
exclusion area	An area where access would result in personnel exceeding radiation exposure limits in a very short time.
Expended Core Facility (ECF)	A large laboratory facility, located at the Naval Reactors Facility in Idaho, consisting of water pools and shielded cells used to receive, examine, and ship naval spent nuclear fuel and irradiated test specimen assemblies. Naval spent nuclear fuel is prepared at ECF for storage and shipment to the Idaho Chemical Processing Plant.
exposure, external	The subjecting of the outside of the body of an organism to ionizing radiation.
exposure, internal	The subjecting of the inside of the body of an organism to ionizing radiation.
exposure, occupational	The subjecting of an individual to ionizing radiation in the course of employment.
exposure, radiation	The subjecting of a material or organism to ionizing radiation.
fauna	Animals.

fissile	A material whose nucleus is capable of being split (fissioned) by neutrons of all energies.
fission	The splitting of a heavy nucleus into two approximately equal parts which is accompanied by the release of a relatively large amount of energy and generally one or more neutrons.
fission products	During operation of a nuclear reactor, heat is produced by the fission (splitting) of "heavy" atoms, such as uranium, plutonium, or thorium. The residue left after the splitting of these "heavy" atoms is a series of intermediate weight atoms generally termed "fission products." Because of the nature of the fission process, many fission products are unstable and, hence, radioactive.
floodplain	The lowlands which adjoin inland and coastal waters and relatively flat areas and floodprone areas of offshore islands which are covered with water from a 1-percent or greater chance flood in any given year.
floodplain/wetlands assessment	An evaluation which consists of a description of a proposed action, a discussion of its effects on the floodplain/wetlands, and a consideration of alternatives.
flora	Plants.
fuel	Fissionable material used or useable to produce energy in a nuclear reactor. It may also refer to a mixture, such as natural uranium, in which only part of the atoms are readily fissionable.
gamma ray	[Symbol γ (gamma)] High-energy, short wavelength electromagnetic radiation. Gamma radiation frequently accompanies beta particle emissions. Gamma rays are very penetrating and are stopped most effectively by dense materials such as lead or uranium. They are essentially similar to x-rays but are usually more energetic and originate from the nucleus. Cobalt-60 is an example of a radionuclide that emits gamma rays.
geology	The study of the origin, history, materials, and structure of the earth.
geophysical survey	An examination of the condition, situation, or value of the earth using the physics of the earth including the fields of meteorology, hydrology, oceanography, seismology, volcanology, magnetism, radioactivity, and geology.

glaciation	The act of having been subjected to glaciers, extreme cold, and ice.
groundwater	Water that exists or flows beneath the earth's surface in the zone of saturation between saturated soil and rock.
half-life, biological	The time required for a biological system, such as an organ or tissue in an organism, to clear by natural (non-radioactive) processes, half the amount of a substance that has entered it.
half-life, radioactive	The time required for half of the atoms of a radioactive material to decay to another nuclear form.
hazardous wastes	Excess chemical material that is dangerous to human health.
health detriment	The sum of all fatal cancers, a fraction of the non-fatal cancers proportional to the severity of the cancer types, and all genetic defects.
health effect	The occurrence of a fatal cancer, a non-fatal cancer, or a genetic defect.
high-efficiency particulate filter	A ventilation system device that can separate a particle size of 0.3 micron from the air into a filter medium at an efficiency of at least 99.97 percent.
hydrology	The study of the properties, distribution, and effects of water on the earth's surface, in the soil and underlying rocks, and in the atmosphere.
incident-free operations	Routine, day-to-day operations without accidents or other unexpected or unusual occurrences. Synonymous and interchangeable with normal operations.
ion	An atom or molecule which has acquired an electrical charge by gaining or losing electrons.
ionizing radiation	Any radiation which displaces electrons from atoms or molecules, thereby producing ions. Examples include alpha, beta, and gamma radiation. Exposure to ionizing radiation may produce skin or tissue damage.
irradiate	To expose to radiation.

isotope	One of two or more nuclides which have the same number of protons but have different numbers of neutrons in their nuclei. Therefore, the isotopes of an element have the same atomic number but different atomic weights. Isotopes usually have very nearly the same chemical properties but somewhat different physical properties.
long-lived radioactivity	Radioactive nuclides which decay slowly, therefore having relatively long half-lives.
man-rem	A unit used to measure the radiation exposure to an entire group and to compare the effects of different amounts of radiation on groups of people. It is obtained by multiplying the average dose equivalent (measured in rems) to a given organ or tissue by the number of persons in the population of interest.
maximally exposed individual (MEI)	A theoretical individual who receives the highest radiation exposure from the facility or activity in question.
maximally exposed off-site individual (MOI)	A theoretical individual located at the point on the DOE site or shipyard boundary nearest to the facility or activity in question.
maximum individual	An individual who could consume items or occupy areas at rates which would be at a maximum for the population of interest.
maximum organ	The organ which receives or could receive the largest amount of exposure to radiation.
metric ton	[Abbreviation MT] A unit of mass which is equal to 1000 kilograms or approximately 2205 pounds.
microcurie	[Abbreviation μCi] A unit of activity which is equal to one-millionth (1×10^{-6}) of a curie.
mil	A unit of length which is equal to one-thousandth (1×10^{-3}) of an inch.
millicurie	[Abbreviation mCi] A unit of activity which is equal to one-thousandth (1×10^{-3}) of a curie.
millirem	[Abbreviation mrem] A special unit for measuring dose equivalents which is equal to one-thousandth (1×10^{-3}) of a rem.

monitoring, environmental	The periodic or continuous determination of the amount of radioactivity or radioactive contamination present in a region.
natural background radiation exposure	The total amount of radiation from cosmic radiation emitted by the sun and the radiation emitted by natural minerals in the earth's crust. Typically, an average annual exposure of 100 mrem to the total body occurs from background radiation.
Naval Nuclear Propulsion Program	A joint program of the Department of Energy and the Department of the Navy which has as its objective the design and development of improved naval nuclear propulsion plants having high reliability, maximum simplicity, and optimum fuel life for installation in ships ranging in size from small submarines to large combatant surface ships. The program is frequently referred to as the Naval Reactors Program.
neutron	An uncharged particle with a mass slightly greater than that of a proton, found in the nucleus of every atom heavier than hydrogen. Neutrons sustain the fission chain reaction in a nuclear reactor.
nuclear disintegration	A spontaneous nuclear transformation which is characterized by the emission of particles and/or energy from the nucleus of an atom.
nuclear fuel	See fuel.
nuclear reactor	A device in which nuclear fission is initiated and controlled to produce heat which is then used to generate power.
nuclear reactor accident	An accident which results in release of fission products from the nuclear fuel.
nuclide	An atomic form of an element which is distinguished by its atomic number, atomic weight, and the energy state of its nucleus. These factors determine the other properties of the element, including its radioactivity.
organ	A group of tissues which together perform one or more definitive functions in a living body.
organism	Any living plant or animal.
overburden	Material overlying a deposit of useful geological materials.
particulate	Pertaining to a very small piece or part of a material.

pathway	The route or course along which radionuclides from defueled nuclear-powered ships could reach man.
percolate	To drain or seep through a material.
permeability	The quality or state of being able to diffuse or pass through a material.
pH	A measure of the relative acidity or alkalinity of a solution. A neutral solution has a pH of 7, acids have pH's less than 7, and bases have pH's greater than 7.
picocurie	[Abbreviation pCi] A unit of activity which is equal to one-triilionth (1×10^{-12}) of a curie.
prototype plants	Land-based naval nuclear reactor plants that are typical of a first design for a naval warship and are used to test equipment and the nuclear fuel prior to use on a shipboard nuclear plant. The prototype plants are also used to train naval officers and enlisted personnel as propulsion plant operators with extensive watchstanding experience and a thorough knowledge of all propulsion plant systems and their operating requirements.
radiation	The emission and propagation of energy through matter or space by means of electromagnetic disturbances which display both wave-like and particle-like behavior. In this context, the "particles" are known as photons. The term has been extended to include streams of fast-moving particles such as alpha and beta particles, free neutrons, and cosmic radiations. Nuclear radiation is that which is emitted from atomic nuclei in various nuclear reactions and includes alpha, beta, and gamma radiation and neutrons.
radiation field	A region where radiation is present.
radiation level	The measured amount of radiation in a region.
radiation survey	The evaluation of an area or object with instruments to detect, identify, and quantify radioactive materials and radiation fields which may be present.
radiation worker	A person specially trained and tested in basic information regarding radiation, its effects, and radiological control techniques and practices.

radioactive contamination	The deposition of radioactive material in any place where it may harm persons, invalidate experiments, or make products or equipment unsuitable or unsafe for some specific use. The presence of unwanted radioactive matter.
radioactive decay	The process of spontaneous transformation of a radioactive nuclide to a different nuclide or different energy state of the same nuclide. Radioactive decay involves the emission of alpha particles, beta particles, or gamma rays from the nuclei of the atoms. If a radioactive nuclide is transformed to a stable nuclide, the process results in a decrease of the number of original radioactive atoms. Radioactive decay is also referred to as radioactive disintegration.
radioactive waste	Equipment and materials which are radioactive and for which there is no further use. Radioactive wastes are generally classified as high-level waste (those resulting from reprocessing reactor fuel or the used reactor fuel itself), as low-level waste, or as low-level waste containing transuranic elements or uranium-233.
radioactivity	The process of spontaneous decay or disintegration of an unstable nucleus of an atom; usually accompanied by the emission of ionizing radiation.
radioisotope	An unstable isotope of an element that decays or disintegrates spontaneously and emits radiation.
radiological consequences	The changes to the environment or the health of a person(s) as a result of the effects of radiation exposure or radioactive materials.
radionuclides	Atoms that exhibit radioactive properties. Standard practice for naming radionuclides is to use the name or atomic symbol of an element followed by its atomic weight (e.g., cobalt-60 or Co-60, a radionuclide of cobalt).
reactor vessel (or reactor pressure vessel)	A very strong, thick-walled steel structure which contains the nuclear fuel and cooling water under high pressure during reactor operations.
rem	A unit of measure used to indicate the amount of radiation exposure a person receives (an acronym for roentgen equivalent man).
risk	The product of the consequences of an event multiplied by the probability of that event.

river stage	The level of the surface of a river in relation to some reference elevation.
sediment	Particles of organic or inorganic origin that accumulate in loose form.
seismicity	The quality or state of shaking or vibrating caused by an earthquake.
shipping container	A specially designed large, stainless steel or lead-lined, steel-shelled cask that is transported in the vertical position on a well-type or depressed center railcar. The container is certified by the Department of Energy and the Department of Transportation for the shipment of naval spent nuclear fuel.
short-lived radioactivity	Radioactive nuclides which decay rapidly, therefore having relatively short half-lives.
socioeconomics	The welfare of human beings as related to the production, distribution, and consumption of goods and services.
special nuclear material	Materials containing nuclides such as plutonium-239, uranium-233, or uranium enriched to a higher percentage than normal in the uranium-235 isotope.
specific activity	The ratio between the amount of radioactive isotope present and the total amount of all other isotopes of that same element, both radioactive and stable. It is usually expressed in microcuries of radioisotope per gram of total element.
specimen	A small sample of material (fuel or non-fuel) inserted into a reactor for testing to characterize the material's performance. Test specimens may be constructed of plant materials, reactor structural materials, or fuel materials.
steam generator	The portion of the nuclear power plant where the heat from the primary system is transferred to the secondary system without physical contact between the water in the two systems.
survey meter	Any portable instrument which is used to detect radiation and is especially adapted for surveying or inspecting an area to establish the existence and amount of radioactive material present.
tectonic	Pertaining to or designating the rock structures which result from the deformation of the earth's crust.
threatened species	Any species or subspecies which is likely to become an endangered species within the foreseeable future throughout all or a significant portion of its range.

GLOSSARY (Cont)

topography	The detailed physical description of the surface of a region, including the relative elevations of features. The graphical representation of the physical configuration of a region on a map.
toxic	Relating to or caused by a toxin which is a poisonous substance that is a specific product of the metabolic activities of a living organism and is usually very unstable when introduced into human tissues.
tritium	A radioactive isotope of hydrogen with atoms that are three times the mass of ordinary light hydrogen atoms. Tritium is present in the reactor coolant as the result of neutron interaction with naturally occurring deuterium present in the water.
uranium	[Symbol U] A natural radioactive element with the atomic number 92 and, as found in natural ores, an average weight of approximately 238. The two principal natural isotopes are uranium-235 (0.7 percent of natural uranium) and uranium-238 (99.3 percent of natural uranium). Natural uranium also includes a minute amount of uranium-234.
vadose zone	The unsaturated region of soil located between the ground surface and water table.
water pools	Deep pools of water that are used to inspect and hold spent nuclear fuel modules. Storage racks are located below the water surface to support and position the fuel modules in place for handling and to prevent the formation of a critical mass.
water table	The upper surface boundary of an uncontrolled aquifer, below which groundwater occurs. It is usually defined by the levels at which water stands in wells that barely penetrate the aquifer.
watershed	The region which drains into a river, river system, or body of water.
wetlands	Those areas which are covered by water with a frequency sufficient to support a prevalence of vegetative or aquatic life that requires saturated or seasonally saturated soil conditions for growth and reproduction. Wetlands generally include swamps, marshes, bogs, and similar areas such as sloughs, potholes, wet meadows, river overflow, mudflats, and natural ponds.

x-rays	Penetrating electromagnetic radiations with wavelengths shorter than those of visible light. They are usually produced (as in medical diagnostic x-ray machines) by irradiating a metallic target with large numbers of high-energy electrons. In nuclear reactions, it is customary to refer to photons originating outside the nucleus as x-rays and those originating in the nucleus as gamma rays, even though they are the same.

ABBREVIATIONS AND ACRONYMS

AEA	Atomic Energy Act
AEC	Atomic Energy Commission
ANL-E	Argonne National Laboratory - East
ANL-W	Argonne National Laboratory - West
ATR	Advanced Test Reactor
Btu	British thermal unit
BWR	boiling water reactor
CAA	Clean Air Act
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFA	central facilities area
CFR	Code of Federal Regulations
cfs	cubic feet per second
Ci	curies
cms	cubic meters per second
CNS	Charleston Naval Shipyard
CWRM	Commission on Water and Resource Management
DEP	Department of Environmental Protection
DOD	Department of Defense
DOE	Department of Energy
EB	Electric Boat Division of General Dynamics
ECF	Expended Core Facility
EDE	effective dose equivalent
EIS	Environmental Impact Statement
EPA	Environmental Protection Agency
ERPG	Emergency Response Planning Guideline
FAA	Federal Aviation Administration
FMEF	Fuels and Materials Examination Facility

ABBREVIATIONS AND ACRONYMS (Cont)

FWPCA	Federal Water Pollution Control Act
HEPA	high-efficiency particulate air
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
IDLH	immediately dangerous to life and health
INEL	Idaho National Engineering Laboratory
INEL-ECF	Idaho National Engineering Laboratory Expended Core Facility
INGL	Ingalls Shipbuilding
KAPL	Knolls Atomic Power Laboratory
KSO	Kesselring Site Operation
kv	kilovolts
kw	kilowatts
kwh	kilowatt hours
LET	linear energy transfer
MCW	maximally exposed collocated worker
MEI	maximally (or maximum) exposed individual
mg	milligram
mgd	million gallons of water per day
MINS	Mare Island Naval Shipyard
MMI	Modified Mercalli Index
MOI	maximally exposed off-site individual
mph	miles per hour
MVA	megavolt amperes
MW	megawatts
MWh	megawatt hours
NAAQS	National Ambient Air Quality Standards
NEA	Nuclear Energy Agency
NEPA	National Environmental Policy Act
NESHAP	National Emission Standards for Hazardous Air Pollutants
NNPP	Naval Nuclear Propulsion Program

NNS	Newport News Shipbuilding
NOAA	National Oceanic and Atmospheric Administration
NOR	Norfolk Naval Shipyard
NPA	nearest public access
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NRF	Naval Reactors Facility
NTS	Nevada Test Site
NYSDEC	New York State Department of Environmental Conservation
OECD	Organization for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
PAH	polycyclic (or polynuclear) aromatic hydrocarbons
PCB	polychlorinated biphenyl
pCi	picocuries
PHNS	Pearl Harbor Naval Shipyard
PHWMA	Pearl Harbor Water Management Area
PNS	Portsmouth Naval Shipyard
PSNS	Puget Sound Naval Shipyard
PWR	pressurized water reactor
RCRA	Resource Conservation and Recovery Act
RWMC	Radioactive Waste Management Complex
SAPS	Shippingport Atomic Power Station
SARA	Superfund Amendments and Reauthorization Act
SNF	spent nuclear fuel
SRS	Savannah River Site
SRS-ECF	Savannah River Site Expended Core Facility
TEDE	total effective dose equivalent
TI	transport index

TLV-TWA	threshold limit value, time-weighted average
TRA	test reactor area
USFWS	United States Fish and Wildlife Service
VOC	volatile organic compound
WIPP	waste isolation pilot plant
WSO	Windsor Site Operation

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